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Proceedings of a Symposium on Advanced
Compact Reactor Systems Held at
Washington, DC on November 15-17, 1982

National Research Council, Washington, DC

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Proceedings of a Symposium
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PREFACE

The National Research Council's Committee on Advanced Nuclear Systems was formed on April 2, 1982, by the Energy Engineering Board of the Commission on Engineering and Technical Systems. Its task was to survey the technology of compact high-energy-density nuclear fission reactor systems and their potential for application to a variety of civilian and military missions. This assessment was sponsored by the Departments of the Army, Navy, and Air Force; the Defense Advanced Research Projects Agency; and the National Aeronautics and Space Administration.

Nuclear power systems were the object of extensive research and development activity during the late 1950s and 1960s. This work was abandoned by the federal government in 1973 largely because the requirements then identified could be met at lower cost by alternative technologies. Today, compact nuclear fission systems are once again being considered by mission planners in civilian and defense agencies. The emphasis is on space applications, including high-powered communications and remote sensing systems, but other possible applications are also of interest.

Since the technology in question has lain largely fallow for the past decade, the committee considered it appropriate to convene a Symposium on Advanced Compact Reactor Systems, to help define the issues and permit experts on the technology to discuss with policymakers and potential users the state of the technology and the range of potential applications. Such a meeting was held at the National Academy of Sciences in Washington, D.C., on November 15-17, 1982. This proceedings is the record of that symposium, containing all unclassified papers submitted by the speakers.

The committee's final report, commenting on the technology and its applications, will be published in April 1983. The committee did not attempt in its report to present a detailed assessment of the technology of high-temperature compact reactor systems; with the time and resources allotted that would have been impossible. The committee was, however, able to survey the technology and provide a framework for structuring an effective research and development program. This,

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the committee believes, is a worthwhile contribution. The communication fostered by the symposium is another. The committee trusts that papers included in this proceedings will stimulate further interest and study by the technical community.

The symposium covered a wide range of subjects, ranging from current technology concepts to safety and regulatory issues. The meeting was well attended and it provided a rare opportunity for the technical community to discuss the many issues surrounding the question of space nuclear power development.

The program was organized by a steering committee composed of Harold Agnew, Robert Avery, Herbert Goldstein, Nicholas Grant, Harold Lewis, Norman Rasmussen, Henry Stone, David Ward, and Robert Wertheim. A great deal of the credit for the success of the symposium, in particular the lively discussion at sessions, belongs to them.

The committee acknowledges the assistance of the staff of the Energy Engineering Board of the Commission on Engineering and Technical Systems. Duncan Brown, the committee's Staff Officer, was responsible for handling administrative arrangements, organizing the symposium, and editing the papers that form the proceedings. Administrative secretaries Julia W. Torrence and Regina F. Dean carried out the word processing, revising and rerevising many hundreds of pages of highly technical documents, with patience and aplomb.

John M. Deutch, Chairman
Committee on Advanced Nuclear Systems

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FOR SPACE POWER

RADIOISOTOPE THERMAL GENERATORS
AND THERMOELECTRIC POWER CONVERSION

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INTRODUCTION

The conversion of thermal to electrical energy by means of the thermoelectric effect has been actively pursued since the early 1950s. Although the physical phenomena that govern this conversion process had been discovered and known since the turn of the last century, practical conversion devices had to await progress in materials sciences before serious consideration for power production could be attempted.

Although all elements, alloys, and compounds exhibit thermoelectric properties, only very few have seriously been considered for the purpose of energy conversion. The reason for this becomes apparent when the basic properties that govern thermoelectric conversion are considered. The usefulness of a material for thermoelectric conversion is dictated by three basic material properties: the Seebeck coefficient, the electrical resistivity, and the thermal conductivity, all of which contribute to make up the so-called figure of merit $Z = \alpha^2/\rho k$.

The conflicting and interdependent requirements of high Seebeck coefficient, low resistivity, and low thermal conductivity impose severe limitation on choice of materials. The Seebeck coefficient was discovered and reported by its namesake Seebeck, in 1822-1823. This work was further augmented by the watchmaker Peltier in 1834. However, it took a hundred years before Joffe, in 1929, first outlined the advantages of thermoelectric generators made of semiconductors. Even Joffe's work was mostly ignored until 1956, when after having been translated it became the foundation for research in the Western scientific world. Though the thermoelectric technology has come a long way since the days when science was made between successive Mondays (the day the Academy of Science would sit in Paris), in many ways it is still a growing technology whose time has yet to come.

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EARLY RTG SYSTEMS: SNAP-3 THROUGH SNAP-11

Radioisotope thermoelectric generators (RTGs) were initially developed as part of the Systems for Nuclear Auxiliary Power (SNAP) program, in which all of the odd-numbered power plants referred to isotope-fueled systems, while the even numbers were reserved for reactor systems. The initial thermoelectric conversion systems reached their peak in the late 1950s and early 1960s. Most of these systems used the "lead telluride technology." This nomenclature was and remains a misnomer. Lead telluride material consists of lead, tellurium, bismuth, tin, antimony, etc. The promise of this power source was being proposed for many serious space explorations. For example, the SNAP-11 system was at one time considered the primary power system for the lunar Surveyor spacecraft series. It utilized a rather short-lived radioisotope, which required considerable power flattening. This was accomplished by a thermal shutter of the isotope fuel cavity that would be open at the beginning of the mission (BOM) and close toward the end of the mission (EOM). An actual generator (SNAP-11) is shown in Figure 1. This generator, electrically heated, was the first of many potential thermoelectric technologies tested at the Jet Propulsion Laboratory (JPL) during the early 1960s. Unfortunately, most of the work in thermoelectric generator development during this period was classified. This ensured continuation of funding but was detrimental to the further development of thermoelectric generators, since it limited the distribution and critical evaluation of the available data.

INITIAL SUCCESSES AND FAILURES

Even as the early application of RTGs resulted in failures, the 1960s saw the first blossoming of thermoelectric generators for space application. Two series of SNAP generators were developed by the Atomic Energy Commission (AEC) and were designated the SNAP-19 and the SNAP-27 series generators. Both of these utilized lead telluride technology. The initial versions of SNAP-19 generators were successfully employed on the Nimbus spacecrafts (weather satellites). A subsequent version of the SNAP-19 was employed by the early Pioneer deep-space missions. This generator was principally developed by the Teledyne Systems, Inc., and is shown in Figure 2. It basically consists of the radioisotope heat source ^{238}Pu contained in a vessel designed to withstand impact reentry (in case of an abortive mission). Since the lead telluride material was not able to be metallurgically bonded to a hot shoe, pellets of N and P legs were pressed through a piston and spring arrangement against the isotope-containing hot frame. Thermal mismatching between the radioisotope and the thermoelectric material required that most of the isotope heat source be surrounded by thermal insulation. Accommodating this thermal mismatch, coupled with the low operating

temperatures and excacerbated by the mechanical contacting
arrangements, resulted in rather low system efficiency.

A modification of this technology provided the power source for the
scientific experiments (ALSEP) left on the lunar landing sites by the
Apollo astronauts. This technology was designated the SNAP-27 series
and was developed by the AEC at General Electric (GE) and the 3M
Company. Figure 3 shows the fueling of this generator by the
astronaut on the lunar surface. The ability to separate the isotope
fuel from the generator during launch, thus providing a separate
reentry vehicle (Apollo 13), was essentially responsible for the
advantages of this system.

DEVELOPMENT DURING THE SEVENTIES

During the 1970s two separate thermoelectric material systems came of
age: a modified lead telluride system and the silicon-germanium
(SiGe) system. The lead telluride system, modified and improved by
the development of a TAGS (tellurium, silver, germanium, tin) segment,
resulted in lower specific-mass systems. Missions such as the Viking
Mars Lander successfully utilized this technology. The lifetime
requirements for this type of mission (1-2 years) enabled these
systems to function at reasonably high operating temperatures
(about 500°C) provided that an inert atmosphere of noble gases (to
reduce the detrimental effects of sublimation) could be maintained
within the generator throughout the mission lifetime. Figure 4
depicts a SNAP-19 generator of this type that was evaluated and life
tested at JPL for the Viking Mars Lander mission.

As noted above, the success of this generator depended on
maintaining hermeticity. A new version of generator systems designed
to operate in the natural vacuum environment of outer space was
developed by General Atomic. This system--Transit--developed for and
used by the Navy, lowered the operating temperatures of the lead
telluride to about 400°C but incorporated a segmented leg of bismuth
telluride for the colder portion. This combination of thermoelectric
materials, coupled by a breakthrough in bonding to the lead telluride
hot leg, resulted in a radiatively coupled system. Figure 5 shows one
of the two generators that were evaluated at JPL for over 4 years of
operation.

The development of the silicon-germanium technology combined the
radiative heat transfer system and high operating temperatures
(1000°C) with vacuum operation. This thermoelectric material had
been under development since the 1960s, yet its operation had been
restricted to the available fuel temperature (about 500°C). The
SNAP-10A is a typical example of the low-temperature SiGe technology
application. The successful metallurgical bonding of this system led
to the development of the "Air-Vac couple" by Radio Corporation of
America (RCA). As implied by its name, it could operate equally well
either in air or in a vacuum. During this time (early 1970s) the

requirements for a long-term-performance conversion system evolved from the Thermoelectric Outer Planet System (TOPS) studies that were needed for the Grand Tour missions, then under consideration by NASA.

In an attempt to eliminate the high cost associated with a specific mission-designed and dedicated RTG, the concept of a multihundred-watt (MHW) RTG was adopted. Its aim was to develop an RTG with an electrical output power of about 100 W. Mission designers could then use one, two, three, or more identical RTGs depending on the specific need of the mission. The basic thermoelectric couple--the unicouple--that became the building block for this generator is shown in Figure 6. The combination of this unicouple concept with the then-designed MHW heat source is shown in Figure 7. Three hundred twelve unicouples (each producing about 0.5 W of electricity) make up a generator producing 150 W at the beginning of a mission. Two experimental spacecraft for the Air Force (LES 8/9) and both NASA/JPL Voyager spacecrafts are powered by this type of generator, which was developed by GE/RCA under the auspices of the Energy Research and Development Administration.

To date, the performance of these RTGs has met every expectation. After more than 4 years of flight operation, the performance of these RTGs is within less than 1 percent of the predicted values (Figure 8), and all indications are for continued normal behavior.

BEYOND THE SILICON-GERMANIUM SYSTEM

Two material systems had been proposed and investigated to improve upon the SiGe system. The first of these, known as the selenide system, was principally developed by the 3M Company. Basically, its promise was an increase in conversion efficiency over that of SiGe. It projected a significant improvement in the mass/power ratio and thus was selected for the principal conversion option for the Galileo spacecraft. As the data base for this technology increased beyond the initial stages, it became apparent that the reduced temperatures that were required to maintain long-term operation lowered the performance of this system to below that of silicon-germanium. Thus its further development was aborted, and the MHW silicon-germanium unicouple, coupled with an improved (modular) heat source, was adopted for the Galileo (as well as the International Solar Polar) mission. The abortive development of the selenide system does point out the necessity of a thorough thermoelectric material characterization and evaluation program prior to committing to a flight developmental program. It is also indicative of the long-term investment required in developing a new thermoelectric material system, which typically takes 10 years.

CURRENT THERMOELECTRIC TECHNOLOGY RESEARCH

The current thermoelectric materials technology research can be classified into two areas: (1) the short-term, modest improvements of existing systems and (2) the long-term, potential gains promised by new material systems.

The first category consists essentially of improvements in the current state-of-the-art SiGe system. This can be accomplished either by increasing the operating temperature or by increasing the basic figure of merit. Figure 9 illustrates the relationship of the figure of merit with conversion efficiency, which can be approximated by the area under the curve. The concept of increasing the figure of merit by modifying the binary SiGe solid solution was first proposed by Syncal Corporation (now TECO) and was actively pursued under NASA/JPL sponsorship. The performance of the current SiGe system is limited by two basic phenomena: (1) the long-range order of the SiGe lattice, which results in a large thermal conductivity due to the lattice contribution, and (2) the limited solubility of dopants within the solution of silicon and germanium, which prevents the free current carrier concentration from being increased to optimal conditions.

The initial investigations showed that additions of gallium phosphide (GaP) did indeed affect both of the parameters in question: The thermal conductivity of the solutions containing GaP exhibited a reduction of thermal conductivity due to increased lattice scattering; also, an increased dopant concentration (decreased Seebeck coefficient) was realized. Although no optimization has been performed, an increase in the figure of merit by about 20-25 percent is a distinct possibility. Added to this improvement of the figure of merit is the use of better vapor suppression coatings and differing geometry and silicon to germanium ratio. This combination could potentially provide for an increase in operating temperature of about $100^{\circ}C$. The success of both approaches will result in a substantial performance improvement over the current systems.

In the second category of technology research is the development of completely new types of materials. These materials, in general do not abide by the conventional or classical behavior for semiconductors. Generally, these material systems fall into one of two categories: the rare earth chalcogenides (La_2S_3 for example) and the boron carbides ($B_{13}C_2$ for example). Both of these material systems exhibit intriguing behavior of thermoelectric properties. These properties have the potential of at least doubling the present conversion efficiencies. Efforts by NASA/JPL have concentrated exclusively on these materials for the past year. DOE also has a small program in this area. It is important to realize, however, that although much can be gained from successful development of these new types of materials, developmental efforts extensive in both time and resources are required to obtain these gains.

THERMOELECTRIC CONVERSION FOR REACTOR SPACE POWER

The use of thermoelectric conversion for reactor space power is novel, and it is, after all, the "only proven" conversion system. This paradox is borne out by the chronology of thermoelectric conversion; the first and only reactor system in the United States (SNAP-10A) utilized thermoelectric power conversion. Yet since these inaugural da__, thermoelectric conversion for reactors has consistently been rejected because of its "low conversion efficiency." For most reactor systems in the intervening years the use of either thermionics or dynamic conversion systems such as Brayton has been proposed. Not until recently has thermoelectric conversion been seriously considered for reactor power. It is still considered a "low efficiency" conversio system; its strength, however, lies in its potential for improvement as well as its being a "low risk" option using current technology.

The current flight-qualified materials technology can be developed to provide a baseline power system. As improved materials do become available, they can be incorporated into the power system without significant design impact. Developing advanced thermoelectric materials provides the potential options for lower system mass. This is shown in Figure 10, which relates the total mass of a reactor space power system to the operating temperatures and technology levels. The use of advanced materials also will result in a smaller, lower volume system, in addition to the mass benefits.

CONCLUSION

Although this paper was by no means intended to be a complete history of thermoelectrics in space, the use of thermoelectric conversion for space power has successfully been demonstrated over the past decades. Increased conversion efficiency and temperature operation are both areas that will lead to improved performance. Both of these areas are presently being pursued through national laboratories and industry.
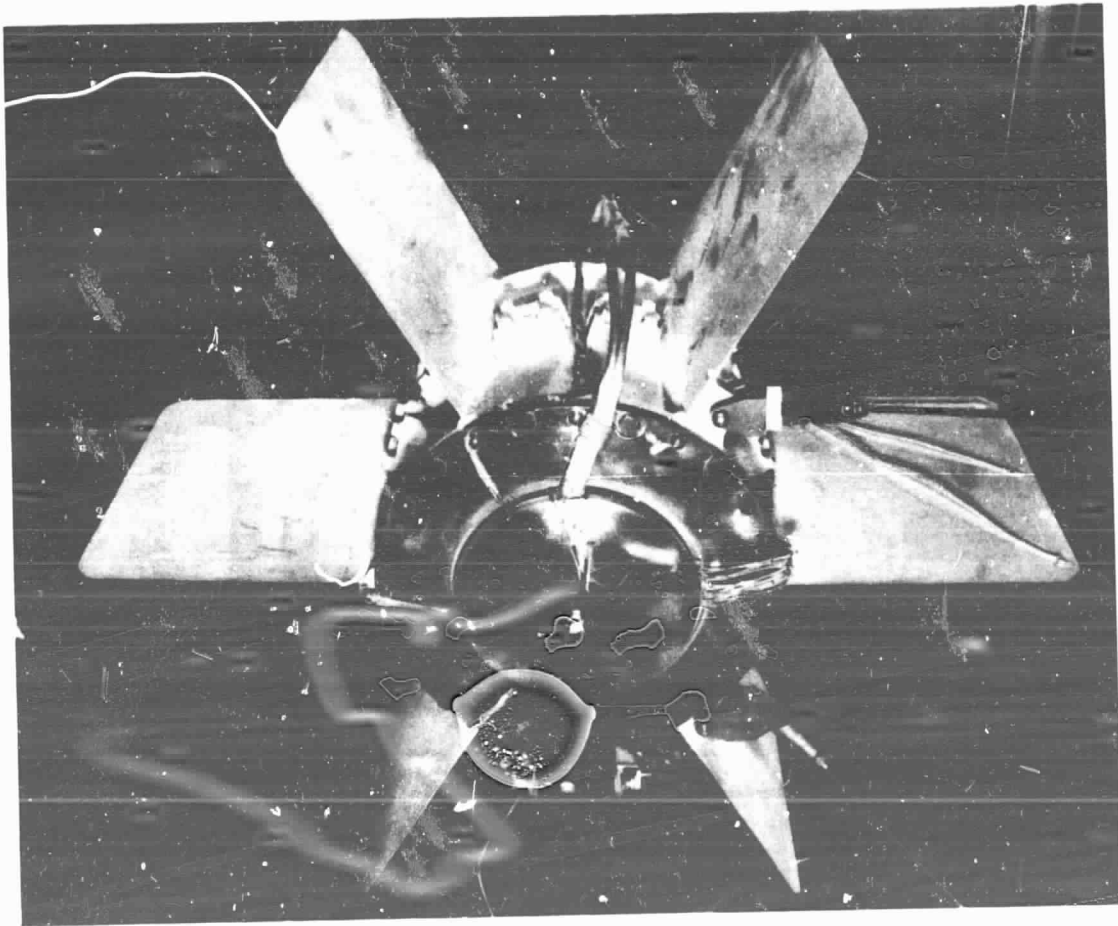
FIGURE 1   SNAP-11 generator.
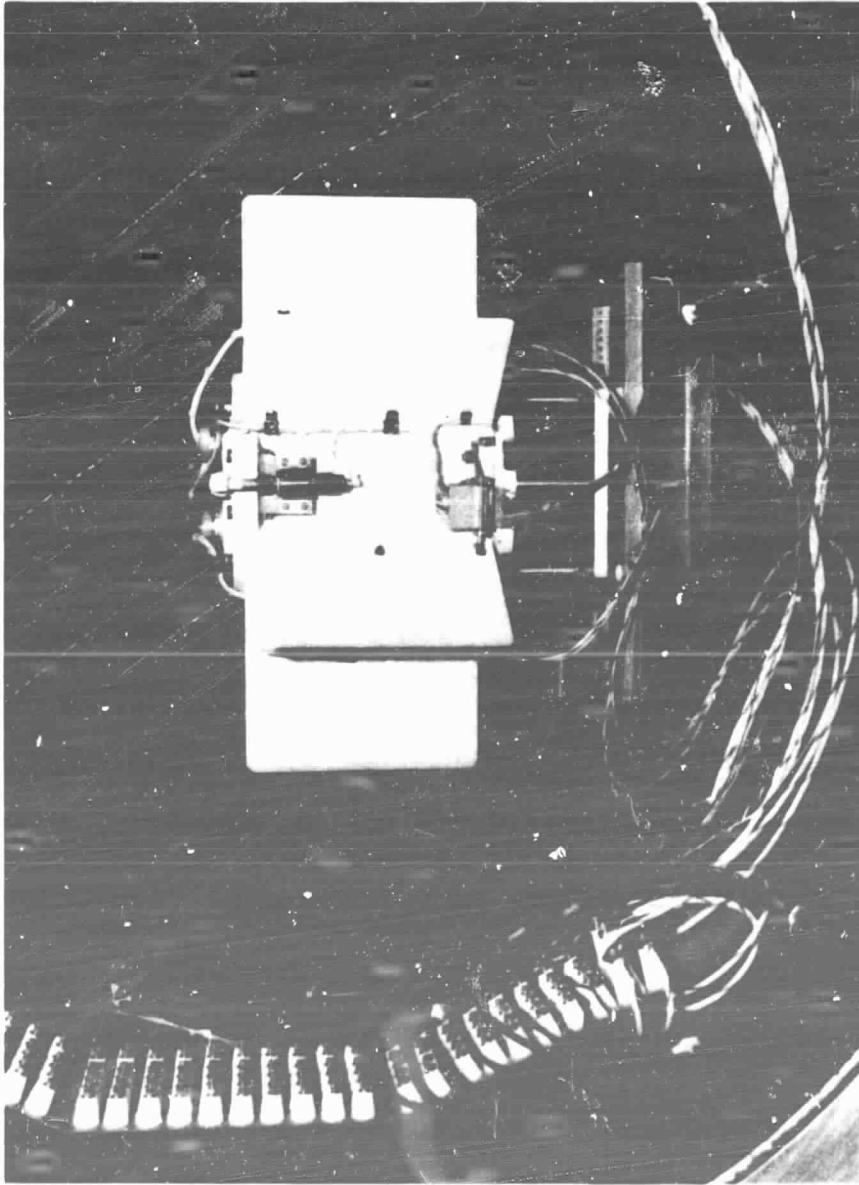
FIGURE 2 Later version of SNAP-19 generator used by early pioneer deep-space missions.
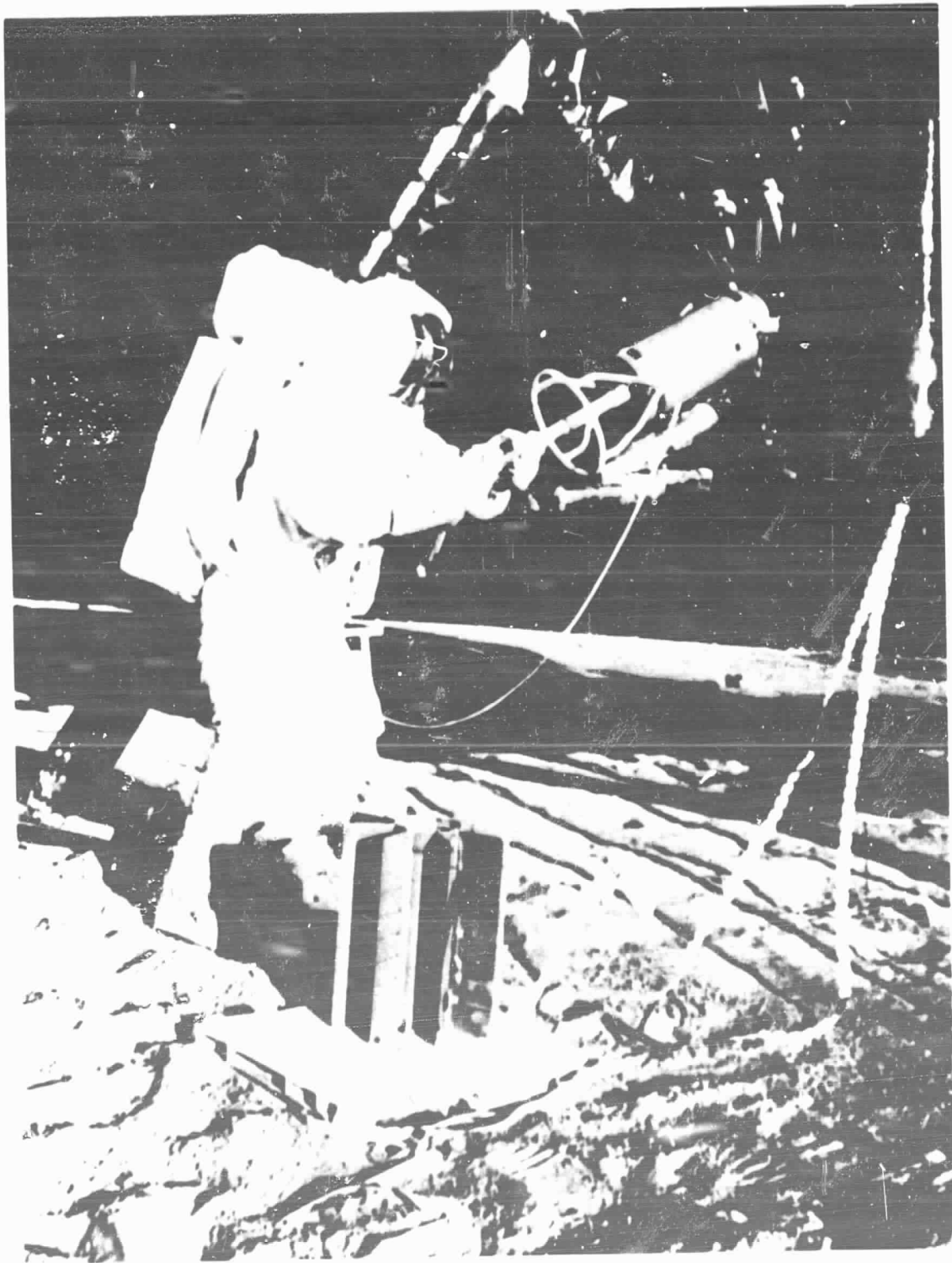
FIGURE 3 SNAP-27 generator being fueled by astronaut on lunar surface.
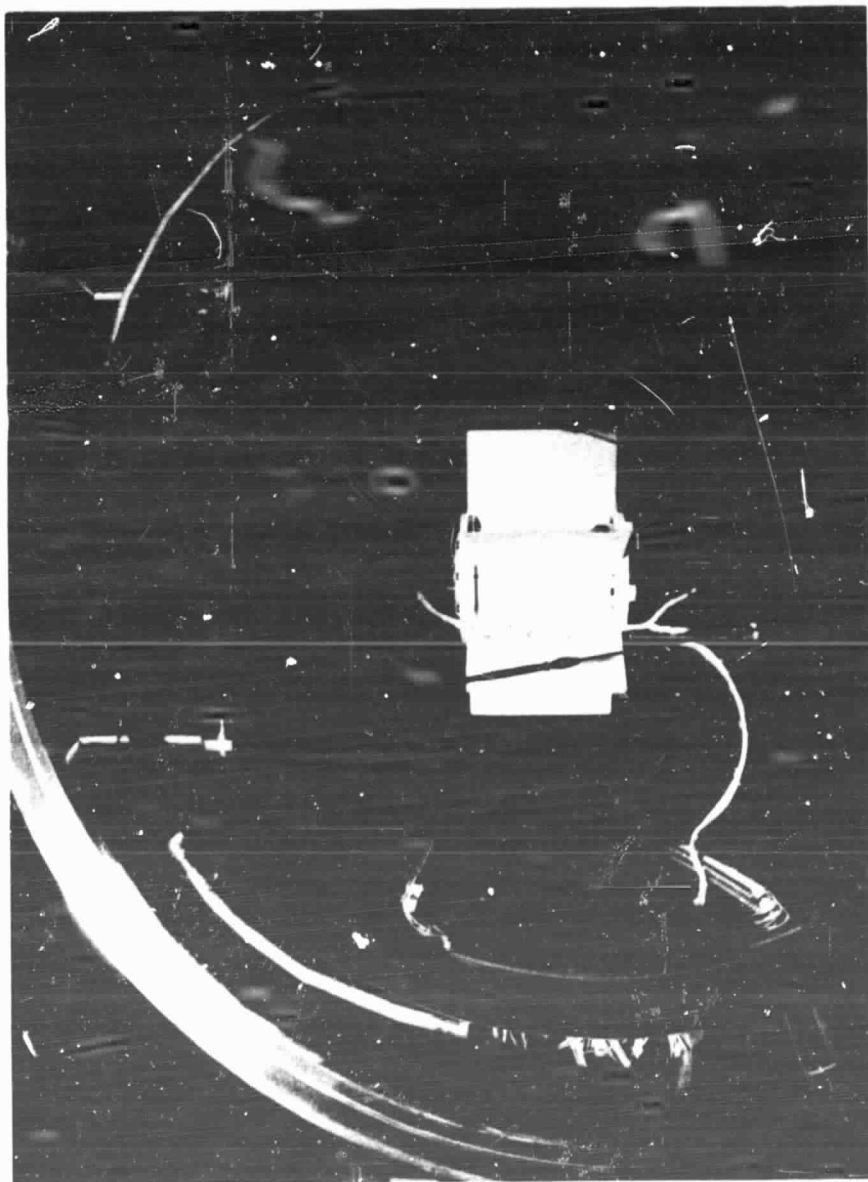
FIGURE 4   SNAP-19 generator using lead telluride technology.
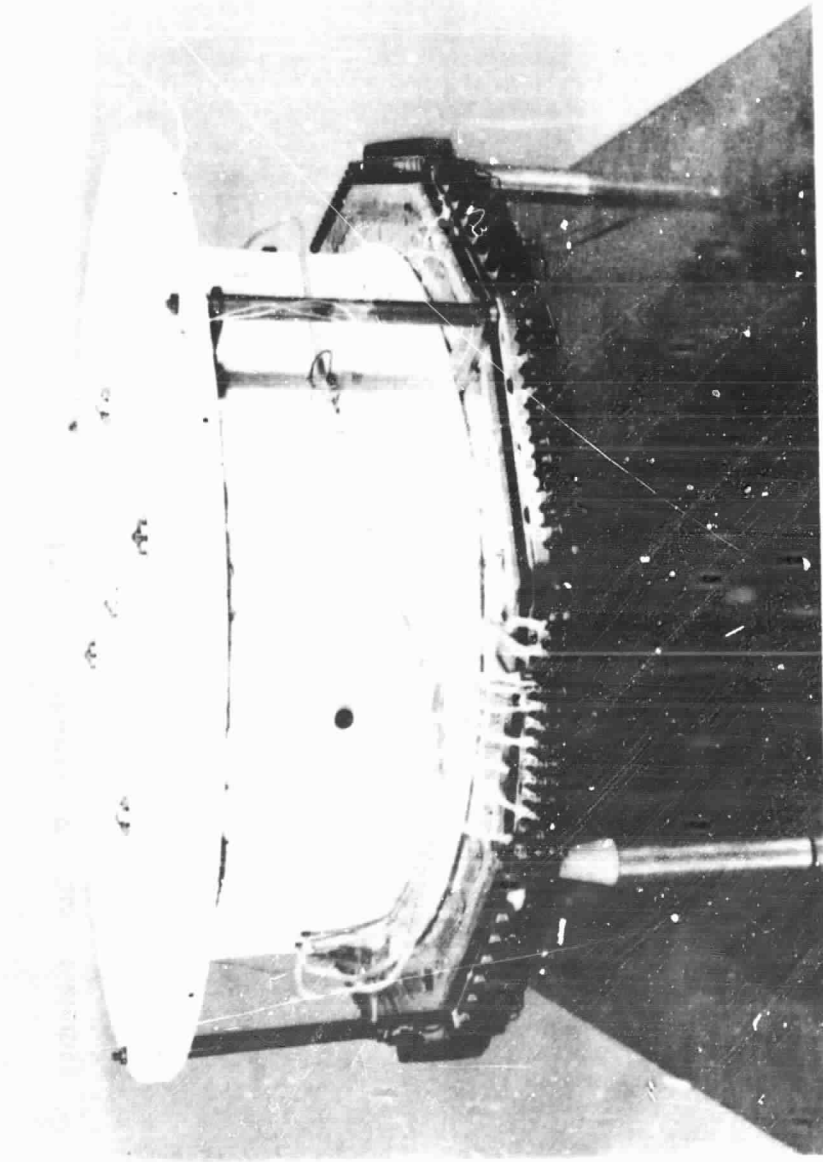
FIGURE 5 Transit generator, a radiatively coupled system.

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SiGe 80% - 20% (Si$_3$N$_4$ COATED) WITH Al$_2$O$_3$ HOTSHOE SPACER

STUFFED WITH MICROQUARTZ, BUT WITHOUT WRAPPING

STUFFED, AND WRAPPED WITH ASTROQUARTZ

METRIC 1 2
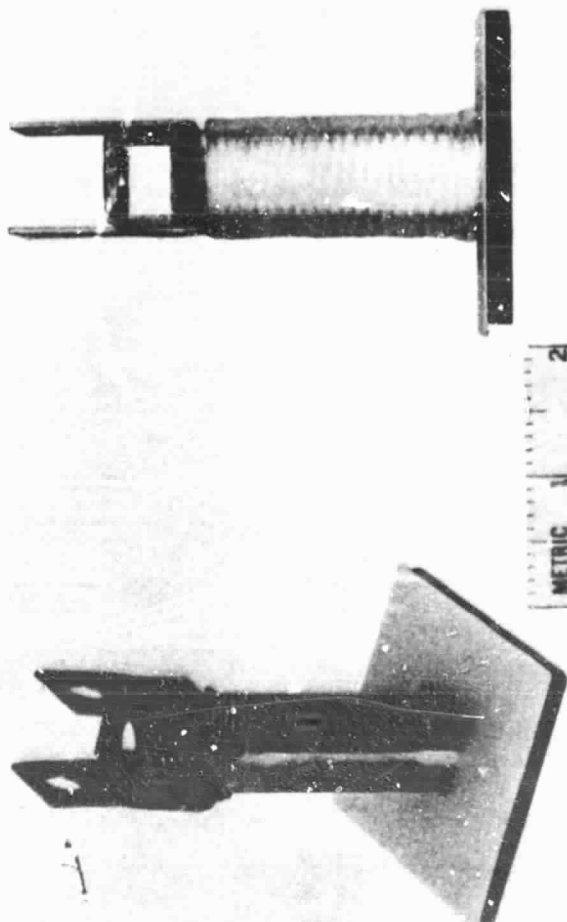
FIGURE 6 RTG unicouples.
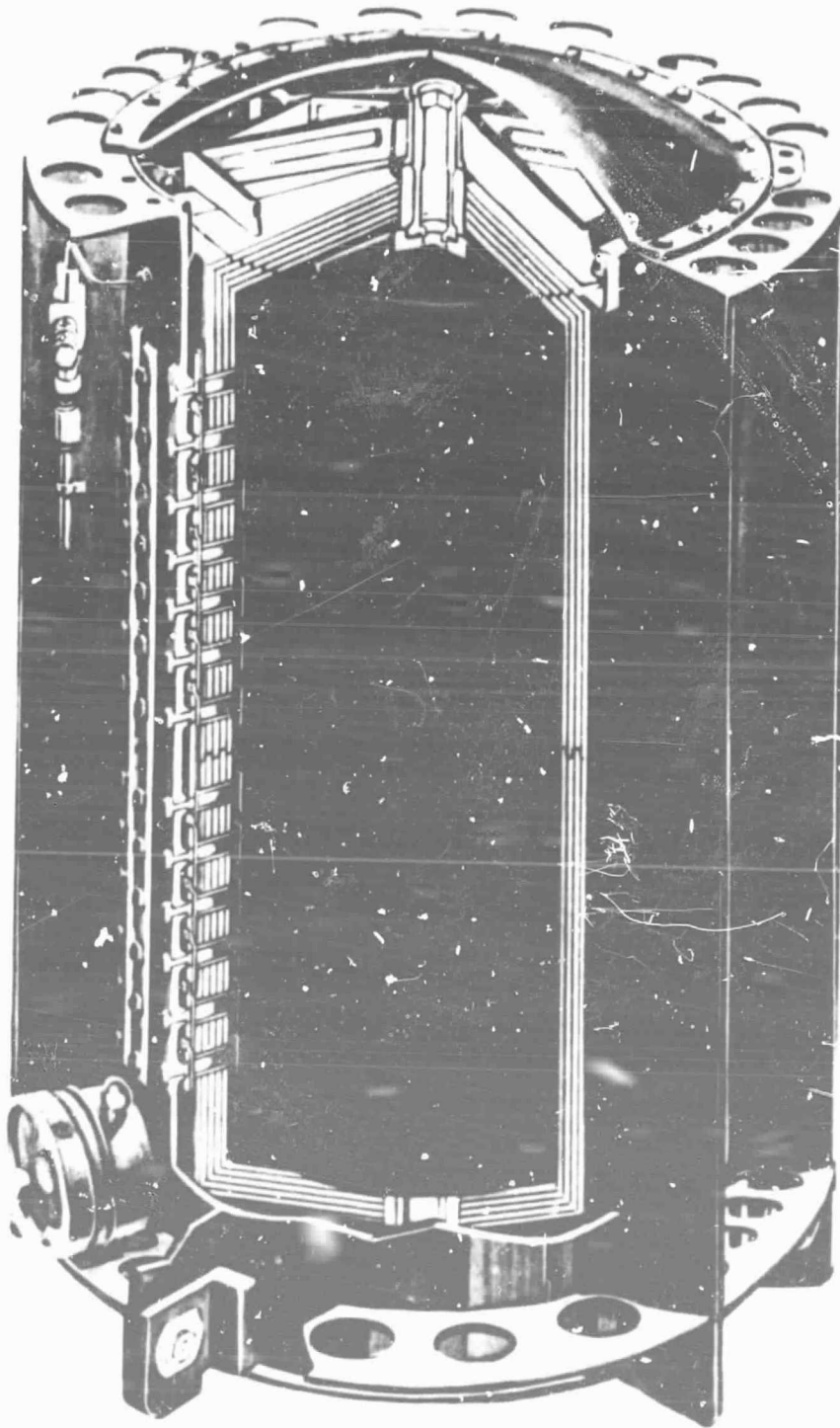
FIGURE 7   RTG using combination of unicouple concept with MHD heat source.

16
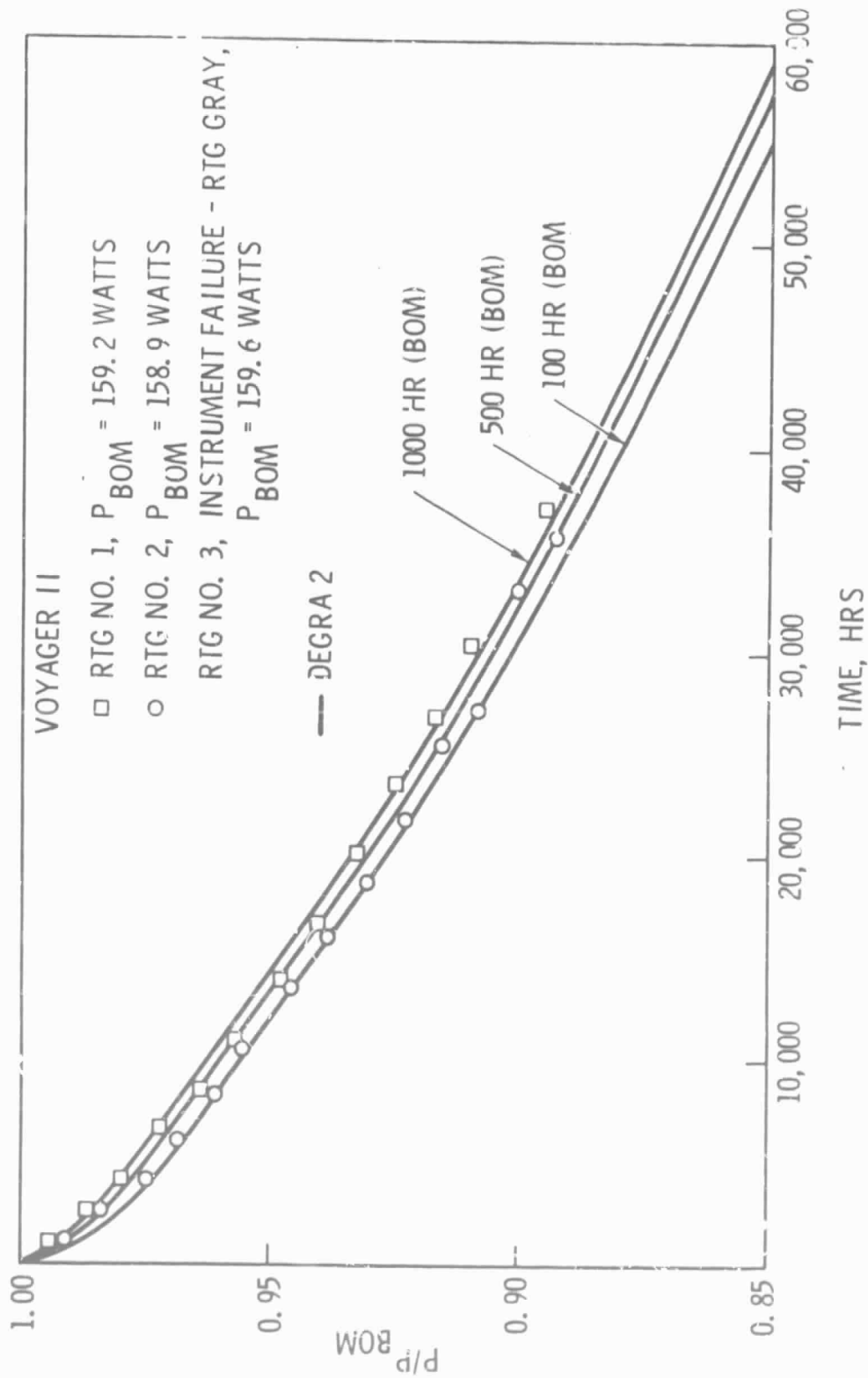
FIGURE 8 Comparison of Voyager 2 Telemetry data and DEGRA 2 RTG performance predictions.
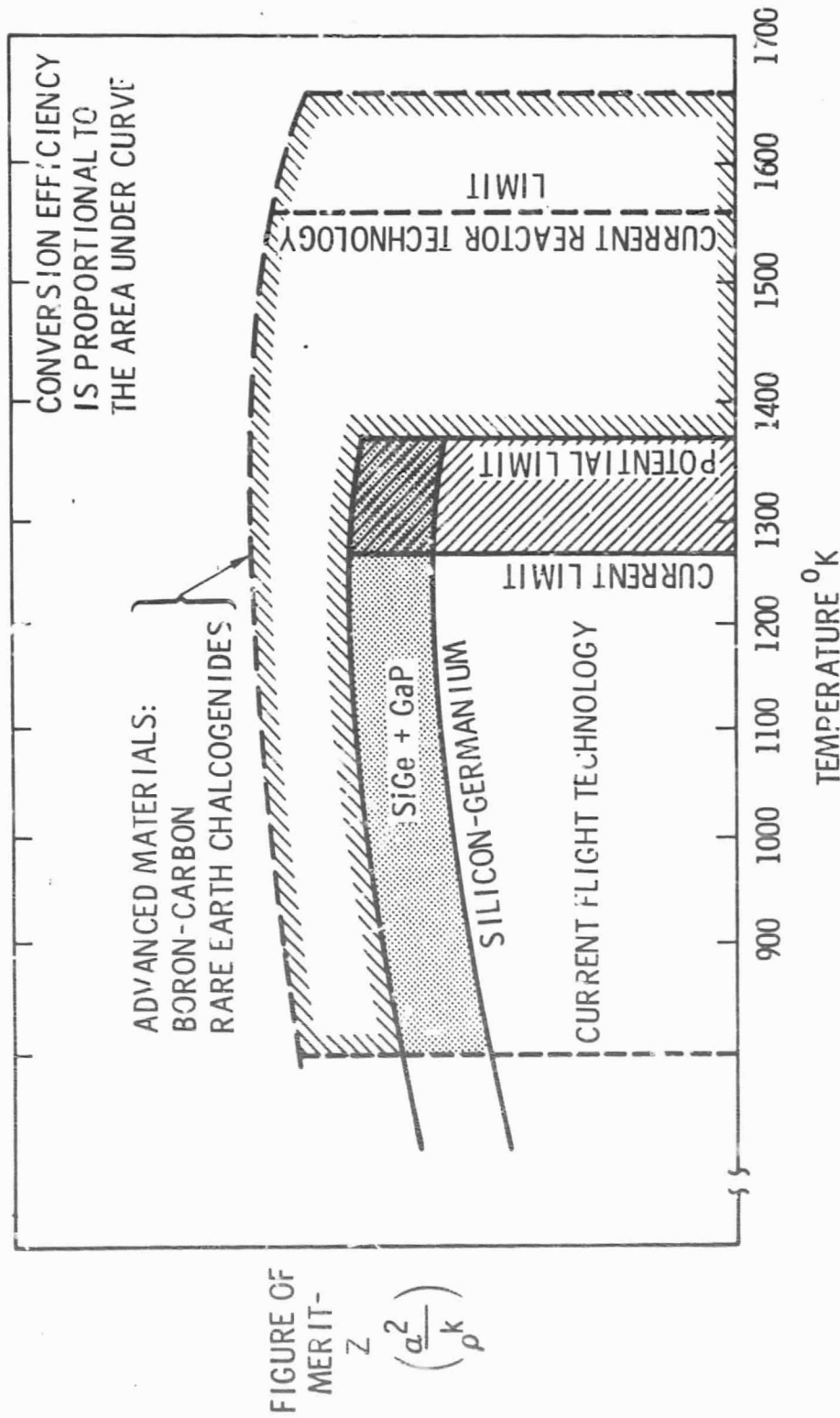
FIGURE 9 Improvement of thermoelectric conversion technology.
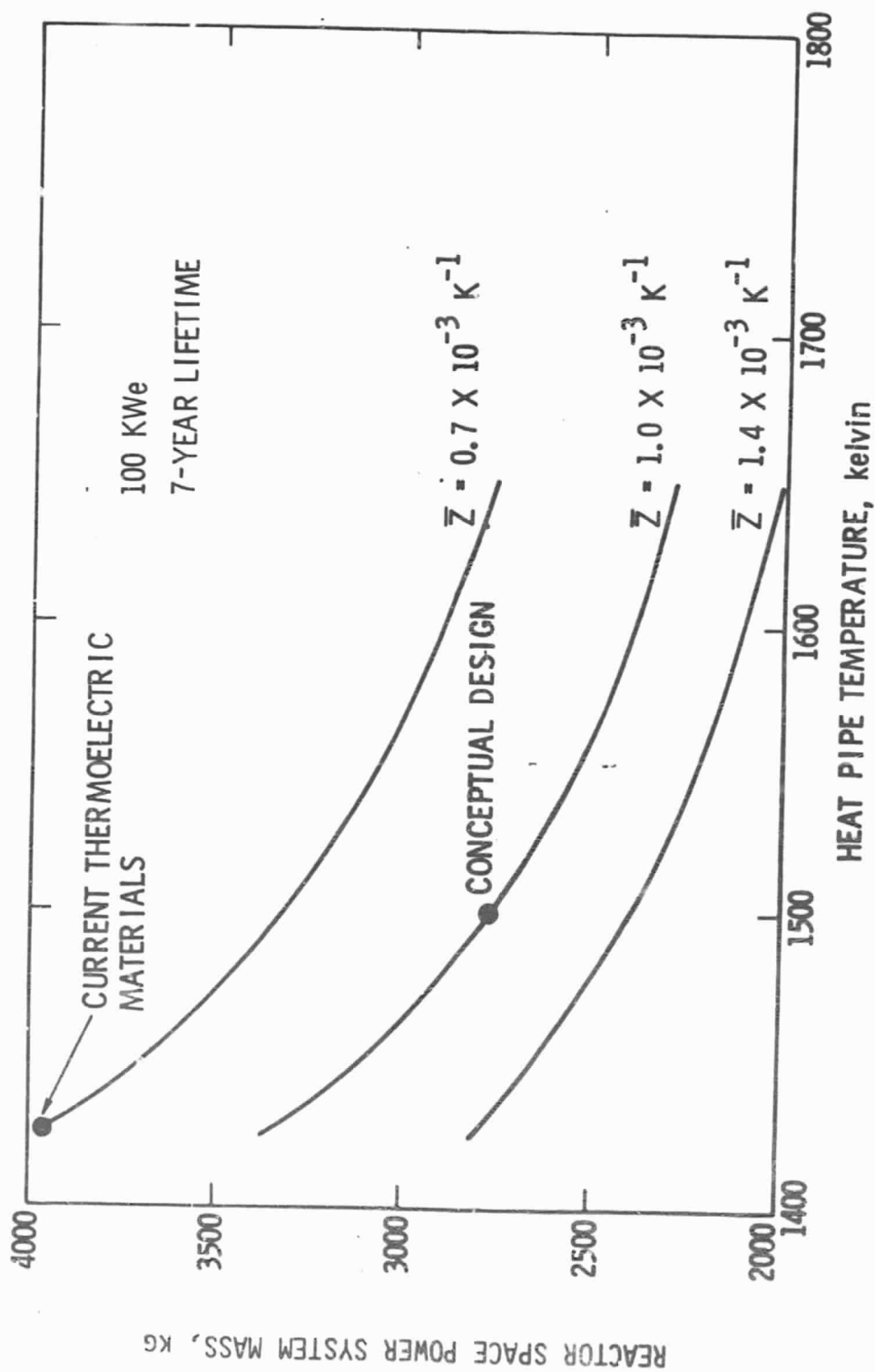
FIGURE 10 Relation of total mass of a reactor space power system to operating temperatures and technology levels.

A NEW GENERATION OF REACTORS FOR SPACE POWER

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ABSTRACT

Space nuclear reactor power is expected to make possible many new
space missions that will require power levels up to several orders of
magnitude higher than that of the largest power source flown in space
to date. Power in the 100-kW(e) range may be required in high-earth-
orbit spacecraft and planetary exploration. The technology for this
power system range is under development for the Department of Energy,
the Los Alamos National Laboratory being responsible for the critical
components in the nuclear subsystem. (The Jet Propulsion Laboratory
is responsible to the National Aeronautics and Space Administration
for the electrical power conversion subsystem development.) The
baseline design for this particular nuclear subsystem technology is
described in this paper; additionally, reactor technology is reviewed
from previous space power programs, a preliminary assessment is made
of technology candidates covering an extended power spectrum, and the
status is given of other reactor technologies.

ESTIMATES OF REQUIREMENTS

Space technologies are being developed at Department of Defense (DOD)
and National Aeronautics and Space Administration (NASA) facilities
for advanced systems of communications, surveillance, electronic
countermeasures, and electrical propulsion. These systems require
tens to hundreds of kilowatts of power. Space nuclear reactor power
plants can meet the needs of potential missions, and, because of
launch weight, survivability, and other considerations, can make the
fielding of practical spacecraft possible. Because associated
spacecraft designs continually evolve, generic missions have been used
to specify space nuclear reactor power plant requirements.

In 1977, generic criteria were specified as the result of a joint
Energy Research and Development Administration (now Department of
Energy (DOE)) and DOD study. These criteria were refined in 1982,
when NASA planetary missions were assessed. Table 1 summarizes these

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TABLE 1 Space Nuclear Reactor Power Plant Design Criteria for 100-kW(e) Systems

Criteria	1977 DOD Requirements	SP-100 Functional Requirements[a]
Power output, end of life (kW(e))	To 100	100
Lifetime (yr)		
Full-power operation	7-10 in geosynchrous	7
System life	3-5 in LEO[b]	10
Reliability		
Design-rated power (percent)	95	TBD[c] (95)
Loss of design power	Favor no single-point failures	No single-point failure
Safety	NASA, ERDA, DOD	Safety guidelines
Survivability		
Nuclear	JCS[d] criteria	JCS criteria
Other		SMATH-d[e]
Radiation at TBD (25 m) distance		
Neutron fluence (nvt)	10^{13} (for 7 yr)	TBD(10^{10}-10^{13})
Gamma dose (rads)	10^7 (for 7 yr)	TBD(10^5-10^7)
Mass (w/o power distribution) (kg)	1,910	TBD (2,000-3,000)

[a]Buden and Stocky (1982).
[b]LEO, low earth orbit.
[c]TBD, to be determined.
[d]JCS, Joint Chiefs of Staff.
[e]SMATH, satellite materials hardening.

criteria; they form the basis for design selection use in the Space Power Advanced Reactor (SPAR) program (a predecessor of SP-100) and the goals for the current SP-100 critical technology program. The SP-100 is being designed to be compatible with the Space Shuttle and associated orbital transfer stages. A continuing activity will further refine these criteria before any commitment to ground engineering system tests is made.

Table 2 presents a summary of DOD potential high-power requirements as presented at the Space Prime-Power Conference in February 1982. Those above the dashed line are planned to be used to refine the SP-100 requirements given in Table 1. Other papers in these proceedings will address some of these missions in more detail. In the next section, we will discuss how the mission requirements influenced our SP-100 design selection and, therefore, how major modifications to the requirements could alter the final decisions on which reactor technology should be pursued.

REACTOR TECHNOLOGY BACKGROUND

An extensive space reactor development program existed in the United States from 1955 to 1973. Since then, research and development efforts have been conducted on a limited basis. These activities are summarized below and in Table 3.

Reactor Core Development

The most extensive fuel development efforts took place during the years 1965 to 1973. The two areas receiving greatest emphasis included uranium-zirconium hydride fuel for the Systems for Nuclear Auxiliary Power (SNAP) program and coated uranium carbide fuel elements for the nuclear rocket (Rover) program. The former qualified fuel elements for 10,000 hours at a 975 K operating temperature using the liquid metal NaK as the core coolant. The latter operated for 2 hours with hydrogen gas coolant at 2450 K. Both were demonstrated in full reactor core tests. In addition, much work was done on other fuel elements. These included uranium oxides, carbides, and nitrides in tests from 2,500 to 12,500 hours. For instance, for reactors cooled by liquid lithium, uranium nitride fuel elements were tested at 1350 K for up to 6,000 hours; gas-cooled uranium dioxide cermet fuel elements were tested at 1800 K for up to 2,400 hours; and thermionic-uranium dioxide fuel elements were tested at 1960 K for up to 12,500 hours.

These various fuel element test results are plotted in Figure 1 and show that a variety of fuel elements covering a range of temperatures appear feasible for 10,000-hour-life systems.

Current space system forecasts predict requirements for reactor cores lasting for 60,000-80,000 hours; the technology base for

TABLE 2   DOD Potential High-Power Requirements

Application	Power Level
Space-based radars	5-400 kW
Surveillance	30-100 kW
Communications	100 kW
OTV (NEP)[a]	100 kW
Jammers	70-200 kW
Lasers	10-100 MW pulsed
Particle beam	10-100's MW pulsed
Advanced concepts	1-100's MW pulsed

[a]OTV, orbital transfer vehicle; NEP, nuclear electric propulsion.

SOURCE: Cohen (1982).

terrestrial nuclear power plants might also be useful. For such lifetimes, heat pipe reactors such as SP-100 may employ uncoated uranium dioxide fuel, while for fluidized bed reactors, coated-particle fuels developed for high-temperature gas-cooled reactors are being considered. These have demonstrated 28,000 operating hours (Parker, 1982).

Coatings for fuel elements act as (1) a chemical barrier between the fuel and the coolant and (2) a physical erosion barrier. A good coating material, besides being chemically compatible with the fuel and coolant, should have good high-temperature stability and match the thermal coefficient of expansion of the fuel. Some coatings are given in Table 4.

Core components such as the support, periphery, and enclosures are difficult to generalize because they usually are unique to a particular design. Gas-cooled reactors usually operate at relatively high pressures and thus require more massive support systems. Liquid-metal-cooled reactors usually operate at low pressure, but corrosion and erosion are more of a concern. Heat-pipe-cooled reactors require a fluid containment structure within each heat pipe and avoid the thermal-hydraulic interaction problems.

Reflector and Control Assemblies Development

Beryllium has been used as the reflector material for a number of space reactors. Various arrangements of movable elements have been used in the reflector for reactivity control. These vary from sliding

TABLE 3 Principal U.S. Space Nuclear Reactor Programs

Power Plant	Purpose	Power Level	Operating Temperature (K)	Period	Type of Reactor	Fuel	Converter	Development Level
Rover	Propulsion	365-5,000 MW(t)	2450	1955-1973	Epithermal	UC	—	Twenty reactors tested. Demonstrated all components of flight engine for more than 2 h. Ready for flight engine development.
Fluidised-Bed Reactor	Propulsion	1,000 MW(t)	3000	1954-1973	Thermal	UC-ZrC	—	Cold flow, bad dynamics experiments successful.
Gaseous-Core Reactors	Propulsion	4,600 MW(t)	10000 / 1500	1959-1978	Fast	Uranium plasma, UF$_6$	— / Brayton	Successful critical assembly of UF$_6$.
	Electricity	4,600 MW(t)	10000 / 1500	1959-1978	Fast	Uranium plasma, UF$_6$	— / Brayton	Successful critical assembly of UF$_6$.
SNAP-2	Electricity	3 kW(e)	920	1957-1963	Thermal	Uranium zirconium hydride	Mercury Rankine	Development level. Tested two reactors; longest test reactor operated 10,500 h. Precursor for SNAP-8 and -10A.
SNAP-10A	Electricity	0.5 kW(e)	810	1960-1966	Thermal	Uranium zirconium hydride	Thermoelectric	Flight-tested reactor, 43 days. Tested reactor with thermoelectrics in 417-day ground test.
SNAP-8	Electricity	30-60 kW(e)	975	1060-1970	Thermal	Uranium zirconium hydride	Mercury Rankine	Tested two reactors. Demonstrated 1-yr operation. Nonnuclear components operated 10,000 h and breadboard; 8,700 h.

TABLE 3 Principal U. S. Space Nuclear Reactor Programs (continued)

Power Plant	Purpose	Power Level	Operating Temperature (K)	Period	Type of Reactor	Fuel	Converter	Development Level
Advanced hydride reactors	Electricity	5 kW(e)	920	1970-1973	Thermal	Uranium zirconium hydride	Thermoelectric and Brayton	PbTe thermoelectrics tested to 42,000 h.
SNAP-50	Electricity	300-1,200 kW(e)	1365	1962-1965	Fast	UN, UC	Potassium Rankine	Fuels tested to 6,000 h.
Advanced metal-cooled reactor	Electricity	300 kW(e)	1480	1965-1973	Fast	Uranium nitride	Brayton and potassium Rankine	Nonnuclear potassium Rankine cycle components demonstrated to 10,000 h. Ready for breadboard loop.
710 gas reactor	Electricity and propulsion	200 kW(e)	1445	1962-1968	Fast	UO$_2$	Brayton	Fuel element tested to 7,000.
In-core thermionic reactor	Electricity	5-250 kW(e)	2000	1959-1973	Fast or thermal driver	UO$_2$, UC-ZrC	In-core thermionics	Integral fuel element, thermionic diode demonstrated for more than 1-yr operation.
Nuclear electric propulsion	Electricity	400 kW(e)	1675	1974-1981	Fast	UO$_2$	Out-of-core thermionics	Limited testing on thermionic elements.
SPAR/SP-100	Electricity	100 kW(e)	1500	1979-present	Fast	UO$_2$	Thermoelectric	Limited testing on core heat pipes and advanced thermoelectric materials.

TABLE 4 Fuel Element Coatings

Fuel	Reactor	Coolant	Coating	Temperature (K)
U-ZrH	SNAP	NaK	Hastelloy N	1035
UC_2	Rover	H_2	NbC, ZrC	2700
UO_2	SPAR/SP-100	Li, Na	Mo-13% Re[a]	2200
UO_2	710	Neon	T-111	1800
			W-30-Re-30-MO	
UN	SNAP-50	Li	Cb-1 Zr-0.090 C	1350
				1500
UO_2, UC	In-core thermionics		W	2200

[a]Heat pipe wall separates fuel from heat transport fluid.

blocks, to rotatable half cyclinders, to rotatable drums with poison segments. All use some form of actuator for movement of the reactivity control element and require bearings in the movable member.

The longest operational times for beryllium reflectors were experienced in reactors fueled with uranium-zirconium hydride. These reflectors were cooled by NaK liquid metal. The SNAP-8 Demonstration Reactor (S8DR) reached a neutron dose of 1×10^{20} nvt and a gamma dose of 1×10^{11} rads (Horton and Kurzeka, 1982). During steady state conditions, the reflector operated at around 600 K. The beryllium was anodized to provide oxidation protection and emittance enhancement. In a 1-year vacuum test, the reflector drive operated satisfactorily with no signs of self-welding or sticking of any component. Bearings of a solid carbon-graphite ball (Figure 2) operated successfully for 7,000 hours in the S8DR reactor test. Four individual bearing sets completed 12,000 hours of vacuum testing at 895 K and 1.3×10^{-3} Pa or lower and altogether accumulated over 100,000 test hours.

Space-qualified, long-life control actuators were most highly developed in the SNAP program. Successful operation of actuators was demonstrated in reactor systems (Donelan, 1973):

Program	Actuators	Hours of Operation
SNAP-10A ground test	2	10,000
SNAP-10A flight	2	1,000
S8DR	6	6,400

The SNAP-10A actuators were designed to provide 8.5 N-cm torque at 615 K; S8DR, 26.1 N-cm of torque with a position of 0.41° at 810 K. In development testing, one of the S8DR units was tested for over 20,000 hours without failure.

Shielding Technology

The longest-life demonstration of space shielding technology is associated with the SNAP program. Similar technology can be applied to a space reactor within the operational constraints of the materials. For SNAP-10A, five cold-pressed lithium hydride (LiH) shields were fabricated. One was used in the SNAP-10A flight and another in the FS-3 ground test. The latter successfully operated some 10,000 hours as part of the reactor test assembly. Visual examination of the LiH block following the completion of the test on March 15, 1966, showed only slight surface darkening and one (out of eight) broken Hastelloy spring. The spring failure was attributed to fatigue. Dimensional measurements of the LiH block revealed that it remained within the accuracy of the measurement (± 1.2 percent). Also, the lattice parameter and density of LiH samples removed from the block were found to agree very well with the unirradiated LiH values (Keshesnian et al., 1973). Activation analysis of the stainless steel vessel indicated fast-neutron fluence of 2.6×10^{18} nvt at the top of the vessel and 8.9×10^{15} nvt at the bottom of the shield. Thermal-neutron fluences of 4.9×10^{17} and 3.8×10^{16} were measured at the top and at the bottom of the shield, respectively. It was concluded that the cold-pressed LiH block withstood the rigors of the reactor experiment without noticeable damage, but improved support was needed to avoid damage during launch.

Evaluation of the methods for fabricating LiH shield shapes led to a melting and casting process instead of the cold-pressing and machining methods. By means of this faster, cheaper, and more versatile process, more structurally reliable shields could be fabricated because the LiH could be solidified in the shield vessel intimately surrounding all internal structural members, penetrations, etc. Also, in SNAP-8, the substitution of lithium enriched with the ^{7}Li isotope reduced the nuclear heating in the shield. This was done in only a segment of the shield to reduce cost. The shield assembly is shown in Figure 3. In the 7,000-hour SNAP-8 test, the LiH shield was exposed to a maximum fluence of approximately 10^{19} nvt and to temperatures ranging from 365 to 500 K. Post-test evaluation showed the shield vessel to be clean and the chromic oxide emissivity coating to be intact, except for some spalling where the top head and the sidewall met. The top of the vessel was estimated to have bulged about 1.6 cm, but no bulging was noted on the sidewalls. Examination of the LiH under the wafer showed the material to be hard and crystalline, as is typical of cast LiH.

THE SP-100 NUCLEAR SUBSYSTEM

On the basis of the above history, analytical studies were performed over a 2-year period to select components for the SPAR power plant in conformance with potential DOD requirements in Table 1. These studies were reviewed following definition of refined requirements for SP-100 and will be reviewed further as more detailed mission studies refine the requirements. The major considerations in selection of a heat pipe reactor for the 100-kW(e) power level were weight, lifetime, and reliability factors. The SP-100 is being designed for 7-10 times the lifetime at one-third the weight of a SNAP-8 power plant and yet with an objective to have no single-point failures.

The SP-100 nuclear subsystem, shown in Figure 4, consists of:

1. the nuclear reactor as the thermal power source,
2. core heat pipes to transport the thermal power from the nuclear reactor to the conversion/radiator subsystem,
3. the radiation shield to attenuate nuclear radiation to the payload, and
4. the reactor controller to regulate the nuclear reactor.

The reactor incorporates heat pipes to transport the heat out of the core to the thermoelectric subsystems. A fast-spectrum reactor fueled with highly enriched UO_2 was selected for long-life, compact core design. Use of UO_2 as the fuel permits operation at high temperatures and maintains material interactions at an acceptable level. Heat pipes were selected to provide many redundant heat removal paths without pumps or compressors to transport heat from the reactor.

The reactor has a core region composed of fuel modules with fuel arranged in layers between circumferential fins, which are attached to the heat pipes. The fins enhance heat transfer from the UO_2 to the heat pipes and thereby reduce the temperatures in the fuel. These fuel modules are in rows around the central safety plug and compose the core. Surrounding the core is a container that provides support to the fuel modules but is not a pressure vessel. The container also provides noncompressive support for the multifoil insulation. Multiple reflective insulation layers lower to an acceptable level the core heat loss to the reflector. The reflector surrounds the core and reflects neutrons back into the fuel region. Located within the reflector are drums that are driven by electromechanical actuators. On part of these drums is a neutron-absorbing material, and the positions of this material are used to establish the reactor power level.

The core heat pipes emanate from the reactor core, bend around the radiation attenuation shield, and continue to the end of the conversion/radiator subsystem, where the heat is transferred by radiation. The radiation attenuation shield is located between the reactor and payload and acts to minimize the reactor-generated

radiation. This shield primarily contains two types of materials--low-atomic-weight atoms (LiH) to attenuate neutrons, and high-atomic-weight atoms (tungsten) to act as a gamma radiation shield.

Figure 5 is a cutaway view of the reactor. The reactor core is made up of heat pipes and fuel modules. The circumferential fins within the core are made of the same metal as the heat pipes, a low-rhenium-content alloy of molybdenum. Fuel tiles are sandwiched between the fins. A central hole in the core provides space for a boron carbide (B_4C) plug that would maintain the core subcritical if water immersion occurred and all heat pipes were broken. This central plug is a safety feature used only during launch and is removed once the shuttle reaches orbit. Figure 6 shows the individual fuel modules. The fins are an integral part of the heat pipe, while the fuel is fabricated as tiles and placed between the fins.

The fuel selected is 93 percent $^{235}UO_2$. As a result, distinct axial and radial power profiles exist in the core. Instead of varying the isotopic content of the fuel, the amount of fuel in each row of modules is varied so that an equal quantity of heat is generated per module. Thus each heat pipe transports approximately the same amount of power. Axial temperature flattening is naturally achieved by the heat pipe itself.

Uranium dioxide fuel swelling is a function of both temperature and time. Data on fuel swelling are given in Figure 7. The design provides void volume to accommodate both the estimated fuel swelling and the thermal stress generated by fuel swelling. Axial swelling is accommodated by the periodic omission of a fuel tile. In the radial direction, void space is provided for fuel swelling.

Failure of a heat pipe to conduct thermal power has led to evaluations of whether excessive fuel temperatures occur. In case of heat pipe failure, transient analysis indicates a peak temperature buildup in a few minutes because of poor thermal contact between adjacent fuel modules. However, within 24 hours the fuel from the failed module is expected to redeposit, increasing the thermal contact and reducing the fuel temperature to acceptable limits.

The reflector reduces neutron losses from the core. The reflector section not penetrated by heat pipes is made of beryllium. The reflector section that is penetrated by the heat pipes is made of beryllium oxide (BeO) and operates at a temperature of about 800 K. It is cooled by radiation to space from its outer surface, and multifoil thermal insulation is wrapped around the core to minimize heat leakage from the core to the reflector. Of most concern is swelling and loss of thermal conductance caused by neutron irradiation. The primary cause of the swelling in beryllium is the formation of helium by (n,2n and α) reactions. At temperatures above 375 K, helium gas accumulates into bubbles, and swelling proceeds as the bubbles grow. No significant swelling (less than 1-2 percent) appears up to approximately 825 K.

Within the reflector are 12 control drums. Each drum has an arc segment of neutron-absorbing material of B_4C on one side.

Reactivity is altered by rotation of the neutron-absorbing material.
Each drum would be rotated with a stepping-motor actuator, which is
located on the side of the radiation shield away from the core for
protection from high nuclear and thermal radiation. The reactor has
built-in excess reactivity (approximately 5 percent) to account for
control margin, fuel burnup, temperature coefficient, and
uncertainties. The B_4C is enriched to 90 percent ^{10}B to provide a
reactivity worth of 12.5 percent. Its volume will increase by about 2
percent after 7 years.

To meet safety requirements, the reflector is designed to
disassemble during atmospheric reentry and consists of segments held
together by molybdenum retainer bands (Figure 8). These bands provide
mechanical integrity to the reactor during launch and operations, yet
rapidly oxidize during atmospheric reentry. Table 5 lists the
characteristics of the nuclear reactor.

Examination of the core heat pipes in more detail shows a closed
containment pipe, a capillary wick structure, the lithium heat
transfer fluid, and a material for gettering impurities. The heat
pipe container is made of a low-rhenium-content alloy of molybdenum,
which was selected because of its potential excellent high-temperature
creep strength and its compatibility with the internal working fluid
and the uranium dioxide fuel. It is ductile at low temperatures, so
that it should be able to withstand the launch loads of the space
transportation system. Lithium is used as the working fluid because
of its high heat capacity and surface tension and its low operating
pressure.

Capillary action returns the lithium from the condenser section to
the evaporator. The artery design provides a low-pressure-drop,
redundant wicking structure. A circumferential distributive wick is
used to move the condensing fluid to and from the arteries, while the
arteries transport the lithium axially. The limitations of heat pipe
performance, in this conceptual system design, are the sonic limit,
which controls the flow rate in the pipe at a given temperature, and
the flow limit determined by the capillary pressure that can be
achieved with a given wick pore size.

Getter materials are included in the heat pipe so that residual and
diffused impurities are absorbed and will not react with the heat pipe
container walls or wick. Typical heat pipe characteristics are given
in Table 6.

The radiation attenuation shield designs are based on the SNAP
program development. The operating temperature of the radiation
shield must be at least 600 K to allow reabsorption of radiolytically
decomposed hydrogen, thus preventing swelling. Conversely, the shield
should remain below 675 K to avoid hydrogen loss from excessive
thermal disassociation if the casing is punctured by meteoroids. To
avoid single-point failures, the LiH is encapsulated in a number of
compartments, so that pressure containment failure will deplete the
hydrogen in only a small part of the shield. The tungsten will
operate satisfactorily at the LiH temperature level.

TABLE 5 Nuclear Reactor Characteristics

Thermal power	1.47 MW
Heat pipe evaporator vapor temperature	1,500 K
Maximum UO_2 temperature	1,730 K
BOL reactivity (K_{eff})	1.05
Fission per 7 yr	8×10^{20} fis/cm^3
UO_2 swelling per 7 year	8%
Burnup	3.6%
Redeposition of Mo and UO_2 on heat pipes	0.5 mm
Core diameter	33.1 cm
Core height	33.1 cm
Be reflector average thickness	9 cm
BeO reflector thickness	10 cm
Reactor diameter	54.2 cm
Reactor height	55.0 cm
B_4C plug mass	3.1 kg
Component masses	
Fuel and fin	185 kg
Heat pipe mass to reactor exit	20 kg
Inner and outer cans, multifoil	9 kg
Fuel module support plates	5 kg
Reflector	150 kg
Actuator	42 kg
Structure	<u>29</u> kg
Reactor total mass (excludes B_4 plug)	440 kg

The radiation shield mass is based on LiH at 94 percent of theoretical density, with 20 percent added to account for such items as structure, casing, loading springs, and reinforcing mesh. The tungsten needs no supporting structure. The characteristics of the shield are presented in Table 7.

The current reactor controller concept employs 12 rotating drums surrounding the core. The control drums will be operated under closed-loop control using functionally redundant controllers. Safety criteria dictate that to prevent overheating of high-temperature system elements, the reactor controller will provide a power cutback. In addition, shutdown control function will scram the control drums if an emergency shutdown situation is detected.

For start-up, temperatures are expected to be controlled on the basis of the temperature of the core heat pipes. To avoid excessive power and temperature overshoots due to long-time constants in transfer functions, the current start-up design takes several hours.

TABLE 6 Core Heat Pipe Characteristics

Number of heat pipes	120
Heat pipe outside diameter	15.5 mm
Wall thickness	0.75 mm
Total heat pipe length	9.08 m
Evaporator length	0.33 m
Adiabatic length	1.22 m
Condensor length	7.53 m
Power transmission design margin	1.5
Design power transmission	12.3 kW
Evaporator radial power density at inside diameter	85 W/cm^2
Evaporator vapor temperature	1500 K
Evaporator wall outside temperature	1507 K
Minimum consensor wall temperature	1480 K
Number of arteries	4
Artery inside diameter	2.1 mm
Distribution screen thickness	0.3 mm
Working fluid	Lithium
Evaporator vapor pressure at 1500 K	83 W-h
Getter material	Hafnium, zirconium
Container ductile-brittle transition temperature	140 K
Heat pipe mass/meter	0.44 kg/m
Total heat pipe mass outside reactor	460 kg

A representative start-up scheme based on methods developed in the Rover program will:

1. ramp the control drums relatively quickly from shutdown to a position well below cold critical
2. ramp the drums through cold critical very slowly
3. switch to temperature control when the temperatures of the heat pipes reach the minimum controllable value
4. raise the power level to increase the control temperature enough to permit normal operation.

Safety is an important part of the control system, and redundancy is the key to providing hardware systems with good safety characteristics. Safety features as now envisioned are as follows:

1. The control drums are pinned in the shutdown position with captive-key locks for all noncritical earthbound nuclear operations.

TABLE 7   Radiation Shield Characteristics

Neutron fluence[a]	10^{12} nvt
Gamma dose[a]	10^6 rads/silicon
Cone half-angle	15^0
Axial thickness	80 cm
Side length	80 cm
Maximum diameter	105 cm
Component mass	
Mass of LiH plus container	485 kg
Mass of tungsten	305 kg
Total shield mass	790 kg

[a]Full power, 7-year accumulation at 25 m from the center of the reactor core.

It will be possible to unlock and operate one drum at a time for testing. The pins will be removed before start-up.

2. A central plug of B_4C is installed in the core for launch and removed after a safe orbit is achieved.

3. The control drum actuator has a brake that holds the drum in position until the brake is energized. The motor does not have enough torque to drive the drum with the brake engaged.

4. With 12 control drums, shutdown could be attained even if some of the drums were to become inoperative.

5. Redundant parallel-operating self-test reactor controllers will ramp the control drums to their shutdown position if either controller fails. The redundant controllers do not share electronic components or energy systems. The system is planned for reset to normal operation and restart after an emergency shutdown.

6. The drums will be spring loaded for return to shutdown in case of loss of electrical power.

The heat from the core heat pipes is radiated to the thermoelectric elements, as shown in Figure 9. Then it is conducted through the thermoelectric material, producing the electrical energy. Insulation is used around the thermoelectric material to reduce the thermal losses. Heat that is not used is radiated from the outside surface to space; this is the cold shoe component of the thermoelectric elements. By distributing the thermoelectric elements over a wide area with a sufficient number of elements, the cold shoe becomes a radiator.

Tables 8 and 9 give the performance and other characteristics of a typical 100-kW(e) power plant. The power plant mass is 2,770 kg if improved silicon-germanium thermoelectric materials are used, and less

TABLE 8   Nuclear Subsystem Performance

Thermal power	1.47 MW
Heat pipe evaporator vapor temperature	1500 K
Component mass	
Reactor	440 kg
Core heat pipe	460 kg
Radiation shield	790 kg
Reactor controller	10 kg
Total subsystem mass	1,700 kg

than 2,000 kg with carbide or sulfide conversion materials. The overall length is 8.5 m for the former conversion.

The current SP-100 power output can be increased by adding rows of fuel modules to the cores and coupling the nuclear subsystem to higher efficiency converters. For example, coupling the current 1,600-kW(t) reactor design concept to a 25 percent efficient Brayton cycle would produce 400-kW(e). Additional rows of fuel modules plus dynamic converter concepts achieve a potential growth to several megawatts (electric).

POTENTIAL TECHNOLOGY CANDIDATES FOR HIGHER POWER LEVELS

As presented in Table 2, several identified potential DOD missions require megawatt power levels. Figure 10 roughly classifies leading technology candidates on the basis of reactor type, conversion system, and heat rejection system. The heat pipe reactor has a potential growth to about 40 MW(t).

As depicted in Figure 10, thermoelectric converters, currently planned for SP-100, are limited to the power-production region below 200 kW(e) because of the number of modules involved and their low efficiency. From 200 kW(e) to the megawatt level, power conversion converters such as Rankine, Brayton, and Stirling cycles would not require any increase in reactor temperatures. Thermionic converters are yet another possibility, but would require reactor temperatures several hundred degrees Kelvin higher. Higher temperature operating reactors would increase fuel swelling and material concerns.

Converter efficiencies of 15-30 percent are possible, but the higher efficiencies lead to lower heat rejection temperatures. Because heat radiated to space follows a fourth-power relationship in temperature (T^4), high reject temperatures lead to reduced radiator areas. As power levels increase, higher heat rejection temperatures

TABLE 9 SP-100 Performance and Mass Summary

	Late 1980s	Early 1990s
Output power (kW(e))		
Range	10-100	10-100
Nominal	100	100
Reactor thermal power (kW(t))		
Range	200-1,600	
Reference design	1,470	950
Design life (yr)		
Design power	7	7
Total	10	10
Overall dimensions		
Length (m)	8.5	7.0
Diameter (max) (m)	4.3	4.3
Radiator area (m^2)	70	43
System mass (at reference design) (kg)		
Reactor and controls	450	380
Shield	790	670
Heat pipes	460	225
TE conversion and circuitry	250	155
Thermal insulation (including end panels)	290	195
Radiator	100	35
Power control subsystems	130	120
Interface equipment and structure	300	205
Total system mass	2,770	1,985
Specific power (W/kg)	36	50

usually dominate the choice of converters. Although work has been performed on these converters in the past, activity on these space conversion systems is presently quiescent. While there appear to be no technology barriers to power plants up to a few megawatts, active development is needed if any of the power options above a few hundred kilowatts are to be available to space mission planners.

As the power level demand expands to tens of megawatts, solid-core, fluidized-bed, or even gaseous-core reactors should be considered.

For space, solid-core reactors were developed most extensively as part of the nuclear rocket program. The Rover design featured a graphite-moderated, hydrogen-cooled core (Figure 11). The 93.15 percent ^{235}U fuel was in the form of uranium dicarbide particles, coated with a pyrolytic graphite. The fuel was arranged in hexagonal-shaped fuel elements, coated with zirconium carbide; each element had 19 coolant channels. The fuel elements were supported by a tie-tube structural support system, which transmitted the core axial pressure load from the hot end of the fuel elements to the core inlet support plate. Surrounding the core was a neutron-reflective barrel of beryllium, with 12 reactivity control drums containing a neutron-absorbing material. The reactor was enclosed in an aluminum pressure vessel. Electric power up to 100 MW could be generated by replacing the rocket thrust nozzle with power conversion equipment. This is a limited-life system, possibly useful for directed-energy weapons (DEWs). A low-power (electric), long-life mode could be achieved by extracting energy through the tie-tube support system.

The Rover technology is ready for flight development, having been tested in some 20 reactors (Figure 12). Peak performances are shown in Table 10.

High-power requirements also might be met by fluidized-bed reactors, in either the rotating or the fixed-bed forms. The former was investigated as a rocket propulsion concept, and the latter has been proposed for space electrical power. A modest research effort in fluidized-bed reactors was carried out from 1960 to 1973.

In rotating-bead reactors, the fuel, in the form of small (100- to 500-μm-diameter) particles, was retained by centrifugal force in a rotating cylindrical structure. The fuel was UC-ZrC and was in an annular arrangement, as seen in Figure 13. A rotor drive unit rotated a cylinder made up of a porous material. Hydrogen propellant passed first through the coolant passages in the rocket nozzle and then through the reflector. All or part of the flow passed through the turbine and then entered the core region. The gas flowed radially inward through the cylindrical structure and annular core at a velocity sufficient to fluidize most of the bed. The heated gas then flowed out through the nozzle. Predicted exit gas temperatures were about 3000 K, with a power density of 1,000 MW(t)/m^3. Control of the rotating-bed reactor was by drums in the reflector that had moderator on one side and neutron absorber on the opposite side. Some typical design parameters are shown in Table 11.

Lower temperature, electric-generation systems can replace the rotating bed with a fixed-bed configuration. The major advantage of the fluidized bed is the simple core structural arrangement. So far, experimental work has been restricted to cold flow tests. Major development is needed to demonstrate a fueled reactor configuration.

Another candidate for megawatt-power reactors is a gaseous-core reactor system. The central component of such a gaseous-core reactor is a cavity where the nuclear fuel is in the gaseous state. The reactor concept shown schematically in Figure 14 is an externally

TABLE 10   Reactor Systems Test Performance

	KIWI-4B8	NRX-A6	Phoebus-2A	Pewee-1
Reactor power (MW)	950	1,167	4,080	507
Flow rate (kg/s)	31.8	32.7	119.2	18.6
Fuel exit average temperature (K)	2330	2472	2283	2556
Chamber temperature (K)	1980	2342	2256	1837
Chamber pressure (MPa)	3.49	4.13	3.83	4.28
Core inlet temperature (K)	104	128	137	128
Core inlet pressure (MPa)	4.02	4.96	4.73	5.56
Reflector inlet temperature (K)	72	84	68	79
Reflector inlet pressure (MPa)	4.32	5.19	5.39	5.79
Periphery and structural flow (kg/s)	2.0	0.4	2.3	6.48

moderated cavity assembly that contains the uranium fuel in the
gaseous phase. For temperature requirements less than a few thousand
Kelvin, the appropriate nuclear fuel would be uranium hexafluoride,
UF_6. Above about 5000 K, uranium metal would be vaporized and
ionized with the fuel as a fissioning plasma. At lower temperatures
it is desirable and at higher temperatures it is necessary to keep the
gaseous fuel separate from the cavity walls. This is accomplished
through fluid dynamics by using a higher-velocity buffer gas along the
wall. Power is extracted by convection or optical radiation,
depending on temperature.

   Gaseous-core reactors offer simple core structures and certain
safety and maintenance advantages. The basic research and development
was completed before program termination, including the demonstration
of fluid mechanical vortex confinement of uranium hexafluoride at
densities sufficient to sustain nuclear criticality.

   Returning to converters, a high-temperature Rankine cycle appears
to offer advantages in closed-cycle, tens-of-megawatts systems because
efficiencies are reasonably high (approximately 20 percent) with high
heat rejection temperature (1000 K). For spacecraft operations,
closed-loop systems appear desirable because of potentially longer
operation times and less interference with spacecraft dynamics. When
radiator size and weight becomes excessive and open-loop systems become
necessary, then a converter like a gas Brayton cycle provides much
higher efficiency than a Rankine cycle.

   An attempt has been made to quantify power plant weights from 100
kW(e) to 100 MW(e). The reactor and shield technologies are deemed
sufficiently developed to make reasonable projections, but converters

TABLE 11 Illustrative Rotating-Bed Reactor Design Parameters

	^{235}U Fuel
Bed internal diameter	63.5 cm
Bed height	63.5 cm
Fuel bed thickness	102. cm
Reflector thickness	
Radial	30 cm
Axial	30 cm
Throat diameter	18 cm
Overall height	123.5 cm
Overall diameter	143.9 cm
Critical mass	156 kg
Bed voidage	60%
Uranium concentration	9.5 at %
Chamber pressure	1,125 psia
H_2 flow rate	20 kg/s
Power (T = 3000 K)	1,000 MW
Reactor weight (including pumps and pressure vessel)	4,750 kg

and radiators above the megawatt range require extensive analysis and development to arrive at realistic values. Figure 15 shows the shift in relative weights in power plants as power level increases. Reactor and shield weights go from being a significant portion of the total system weight in the 0.1- to 1-MW(e) range to only a small percentage of the total at 100-MW(e). At 1 MW(e), the converter is the dominant weight component. Above this level, the radiator dominates. Major advances will be needed in converter and radiation technologies to develop practical closed-loop power plants in the 100-MW(e) range.

SUMMARY

The SP-100 class of nuclear subsystem design provides a basis for critical component development. As mission requirements become better defined, as more detailed analyses are performed, and as the results of component testing are completed, this particular design can be updated. The heat pipe concept appears to be a reasonable baseline. Plans are being considered to seek improvements both in the general reactor design approach and in specifics on component details.

The SP-100 technology may be capable of meeting power levels 4 to 8 times that of the largest power source flown to date (25-kW(e) in Skylab) if actively supported and carried through to flight

development. Extension of SP-100 technology might meet power requirements 50 times this level. At higher power levels, other versions of nuclear power plants must be investigated and detailed studies must be done to identify the best technological approaches.

Those realistic reactor concepts that can meet high-priority mission needs will, of course, have near-term priority. If necessary, some limitations in performance may be accepted to accelerate availability. As a first step, a good system study to evaluate technology candidates for the 1- to 100-MW(e) regime is required. It could identify the prime candidate for a reactor technology for up to 10 MW(e), and possibly a candidate for a reactor technology for up to 100 MW(e). Converters might include a midrange closed-loop system, such as a Rankine cycle, and a high-range open-loop system, such as a Brayton cycle. A modular approach might minimize the number of unique units to be developed. Heat rejection systems will require definite technology advancements for the higher megawatt regime.

Mission requirements and reactor and conversion subsystem technologies must be assessed early prior to embarking on a vigorous national space reactor power system development.

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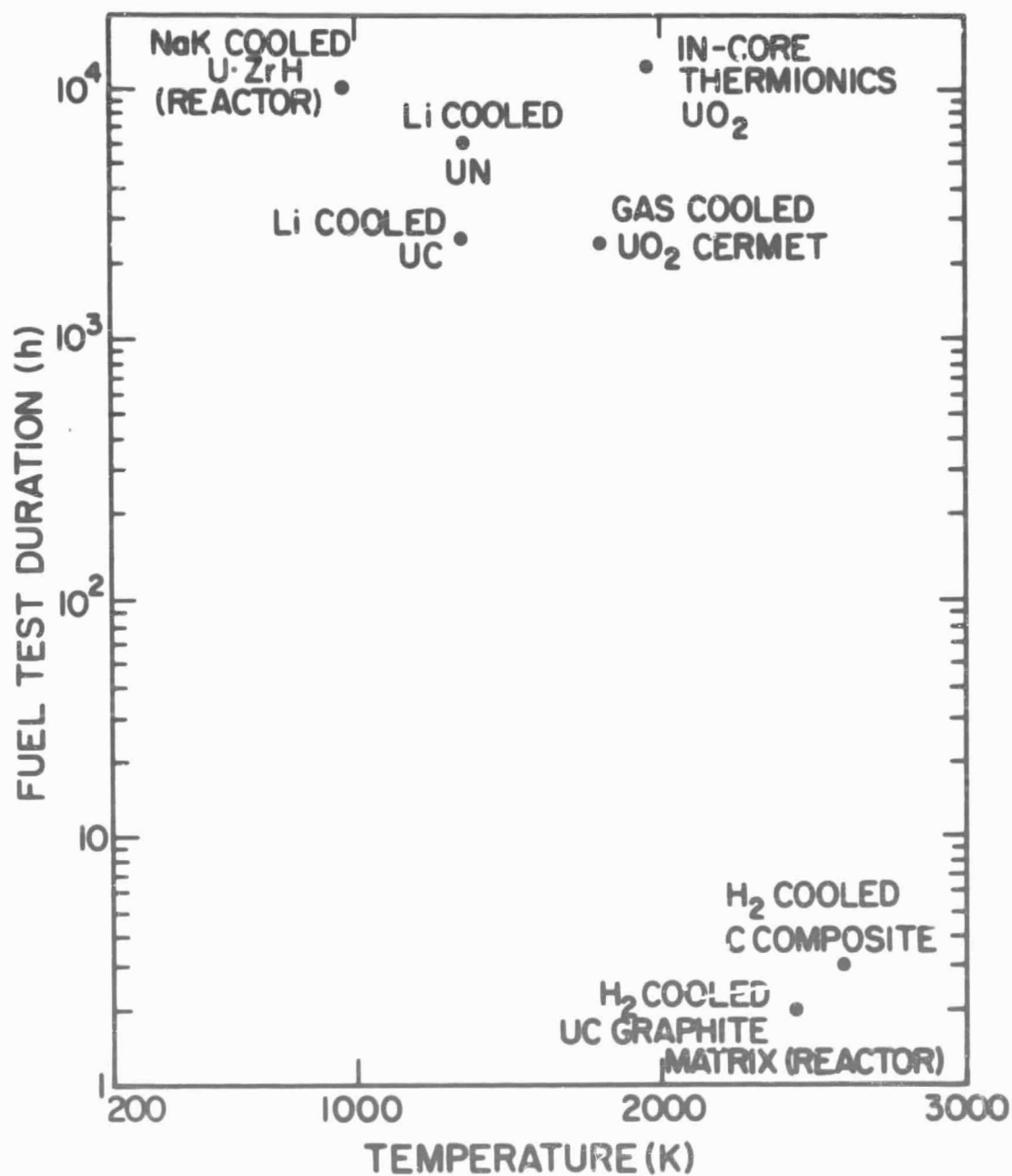
FIGURE 1 Fuel technology (maximum-duration single test).
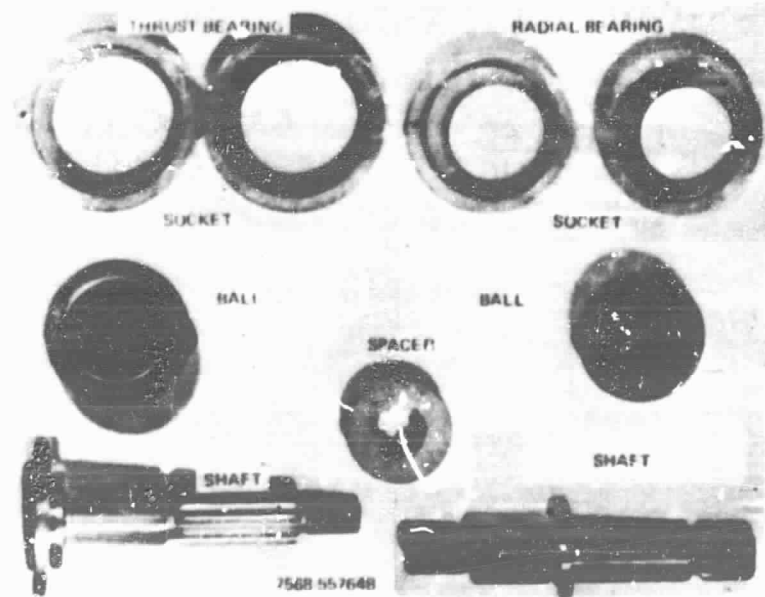
FIGURE 2   SNAP-8 composite bearing.

SKIN THERMOCOUPLES WITH GOLD
PLATED STAINLESS STEEL OVERLAY

TRANSDUCER ASSEMBLY

ALUMINUM FOIL-BACKED LO-CON
XSWF INSULATION
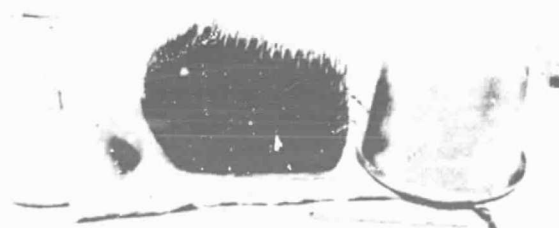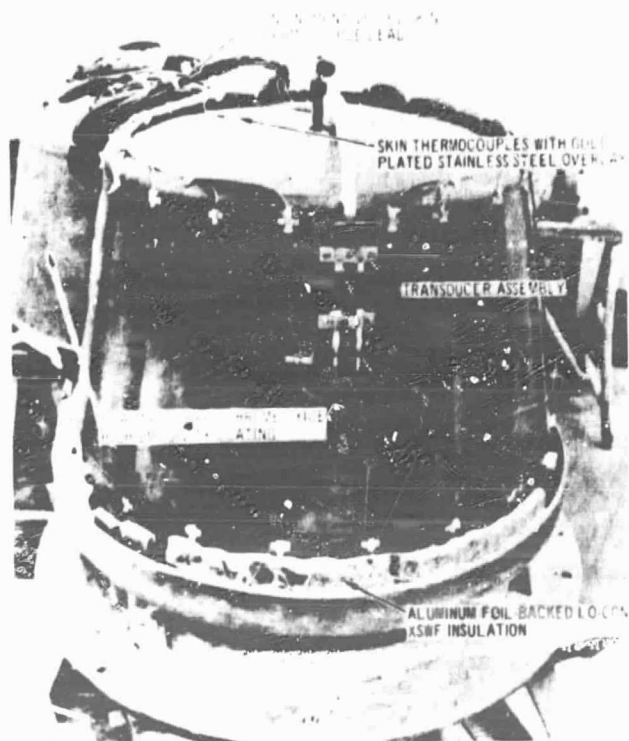
Shield Vessel, Shaped Honeycomb, and Vessel
Bottom with Filler Tube and Bellows Attached

FIGURE 3   S8DR shielding assembly.
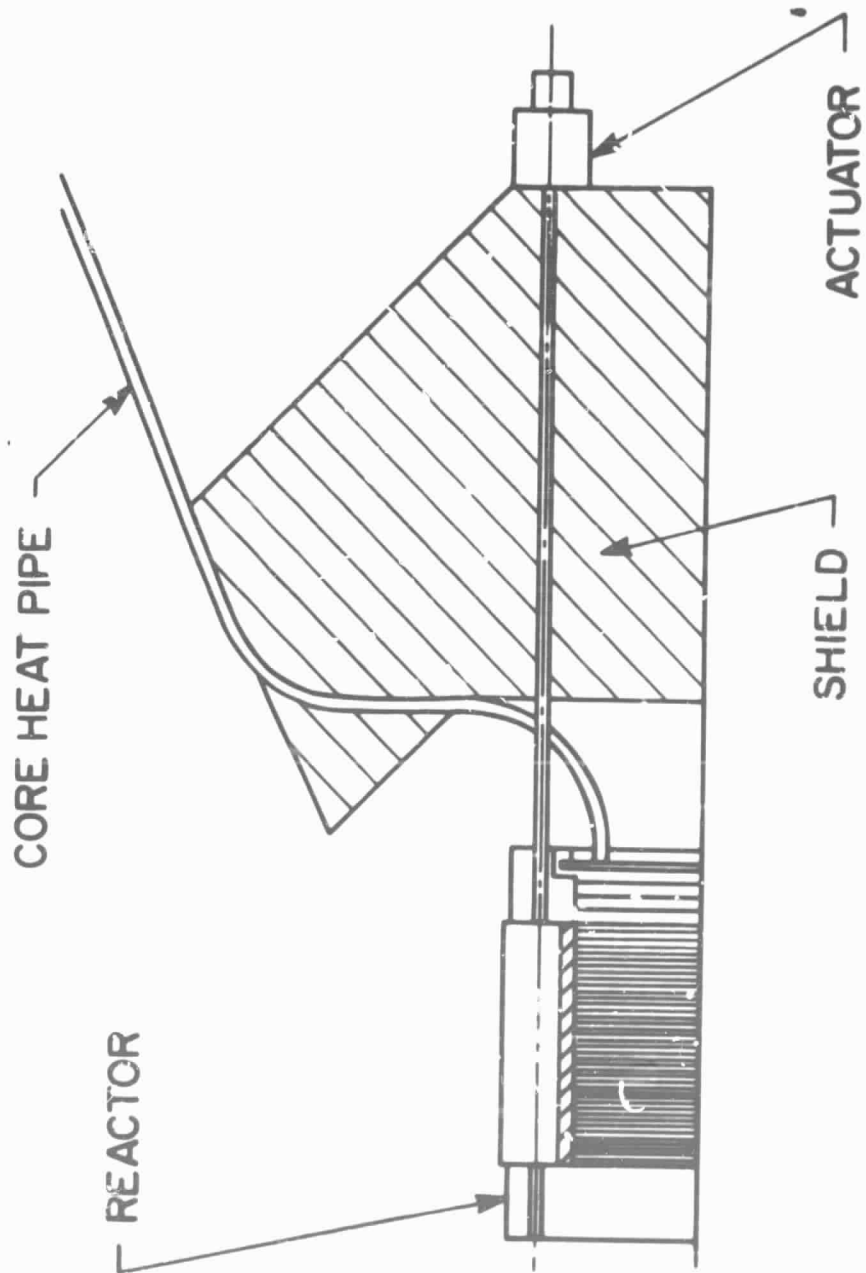
FIGURE 4   SP-100 nuclear subsystem.

43
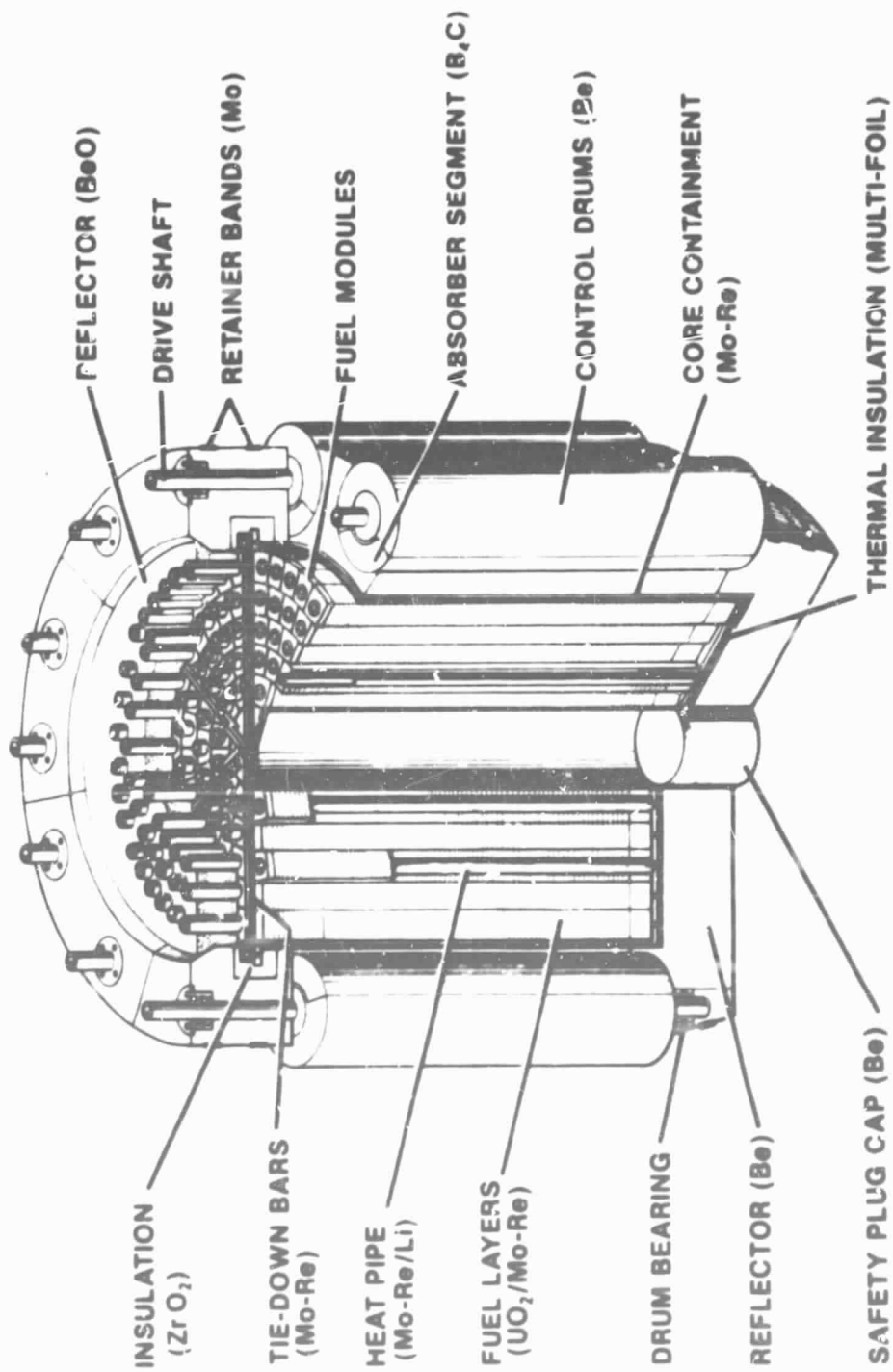
FIGURE 5   Heat pipe space reactor.

STRESS RELIEF SLOT

FIN

UO$_2$

BeO
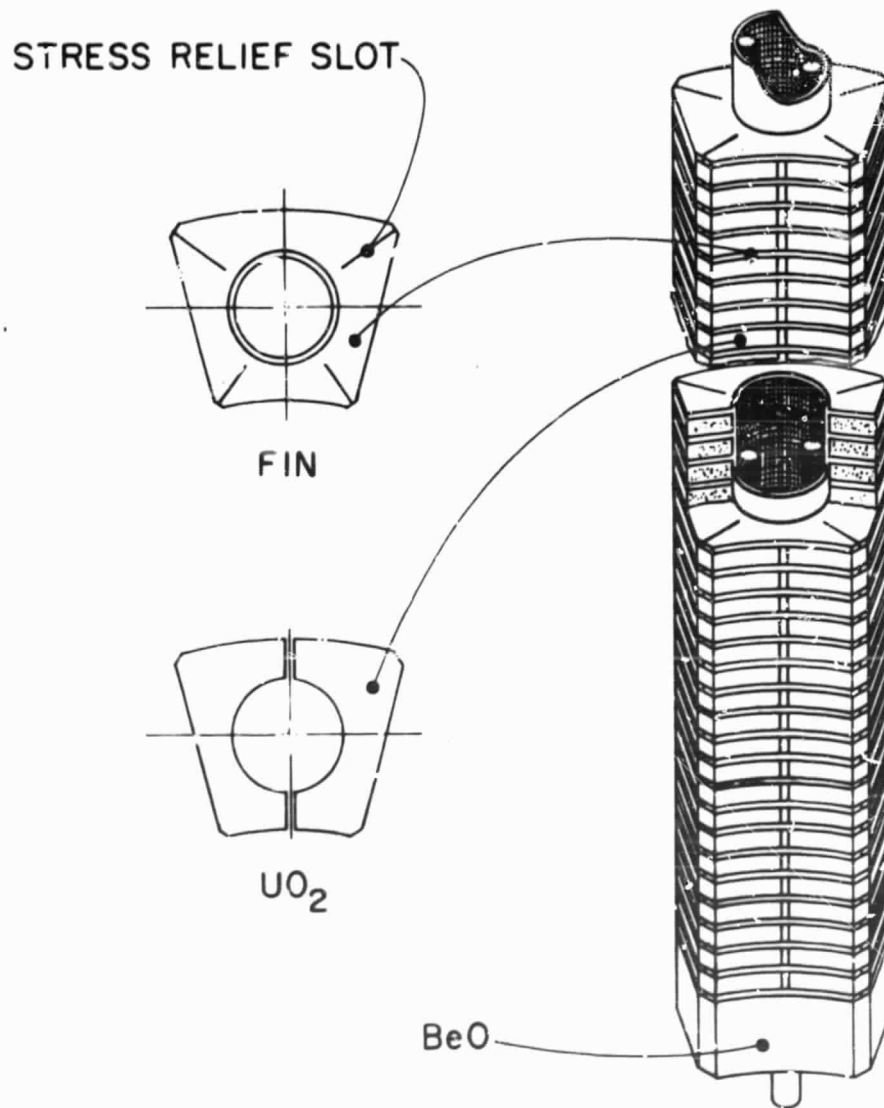
FIGURE 6   Heat pipe fuel module.

45
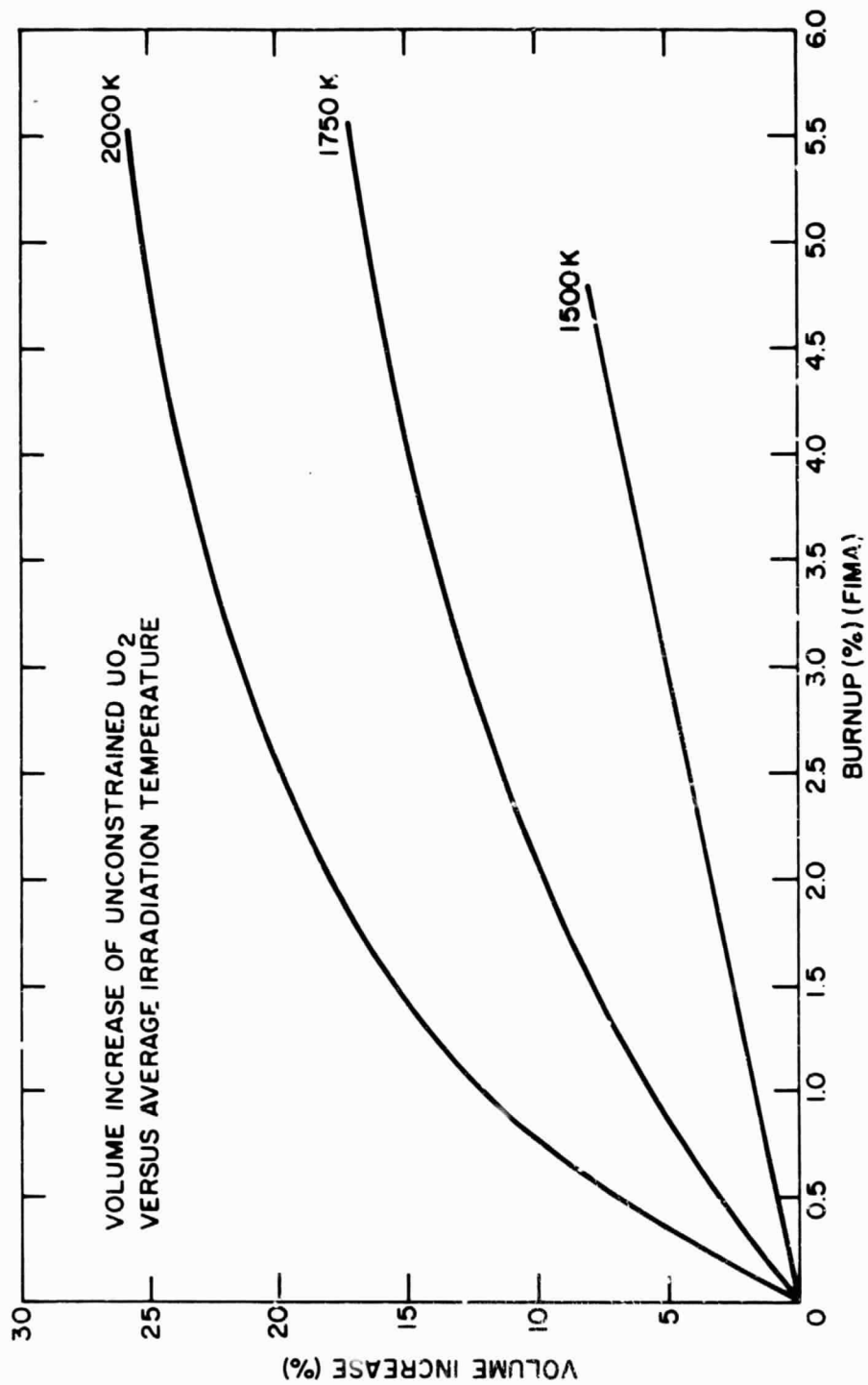
FIGURE 7 Fuel swelling.
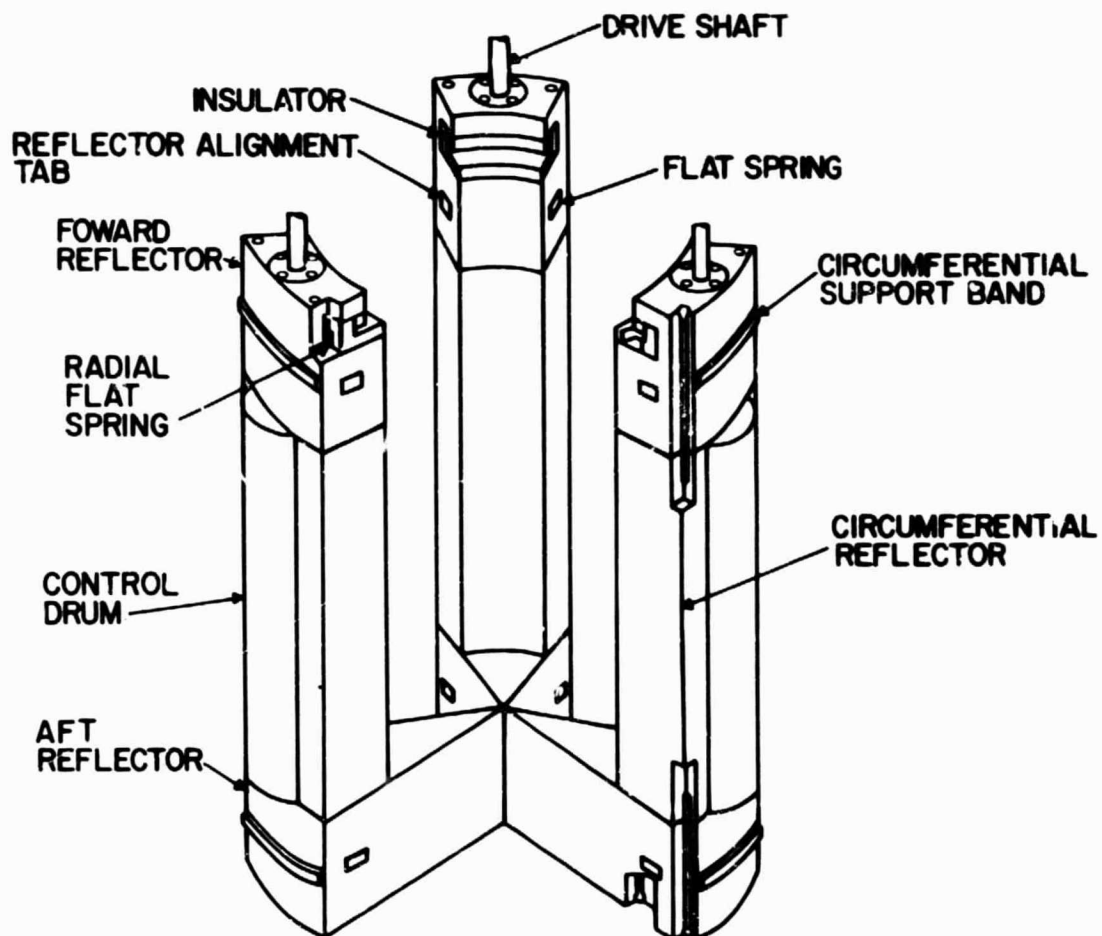
FIGURE 8 Reflector segmentation.

47
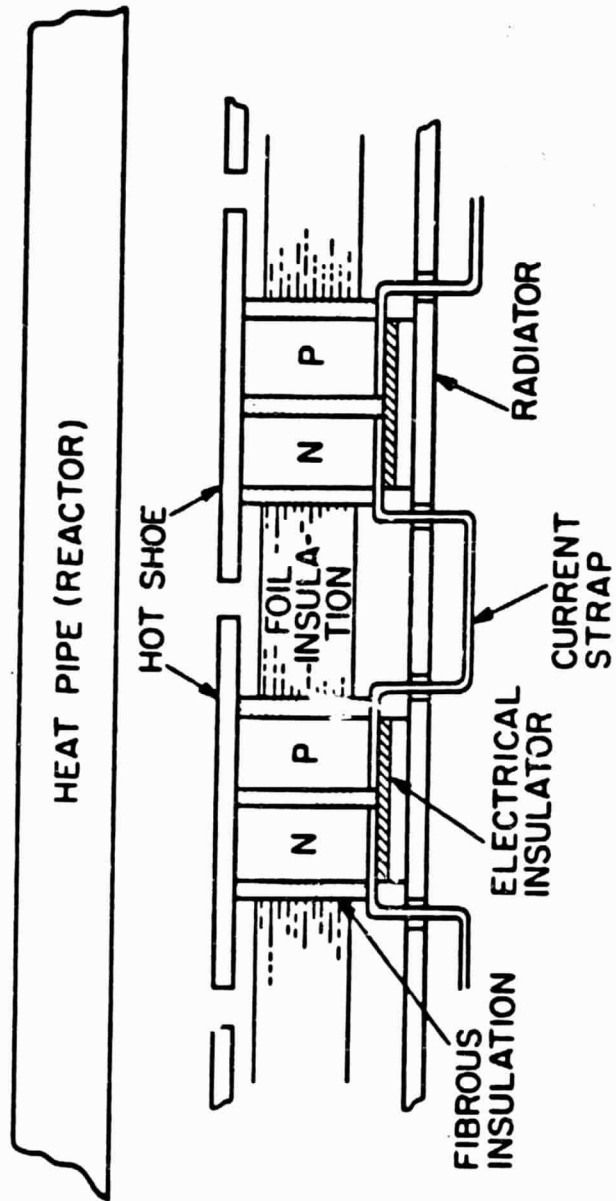
FIGURE 9 Unicouple/panel concept.

REJECT HEAT (AREA m²)

CONVENTIONAL TECHNOLOGY

80

CONVEN-
TIONAL
ADVANCED*

ADVANCED?
500

SPECULATIVE
OR OPEN LOOP
5000

* Liquid Droplet, Ribbon

CONVERTER

THERMOELECTRICS

THERMIONICS
BRAYTON
RANKINE
STIRLING

RANKINE

BRAYTON
(OPEN
LOOP)

MHD

REACTOR

HEAT PIPE

HEAT PIPE
SOLID CORE

SOLID CORE
FLUIDIZED BED
GAS CORE

POWER (MWe)

0.01   0.1   1   10   100
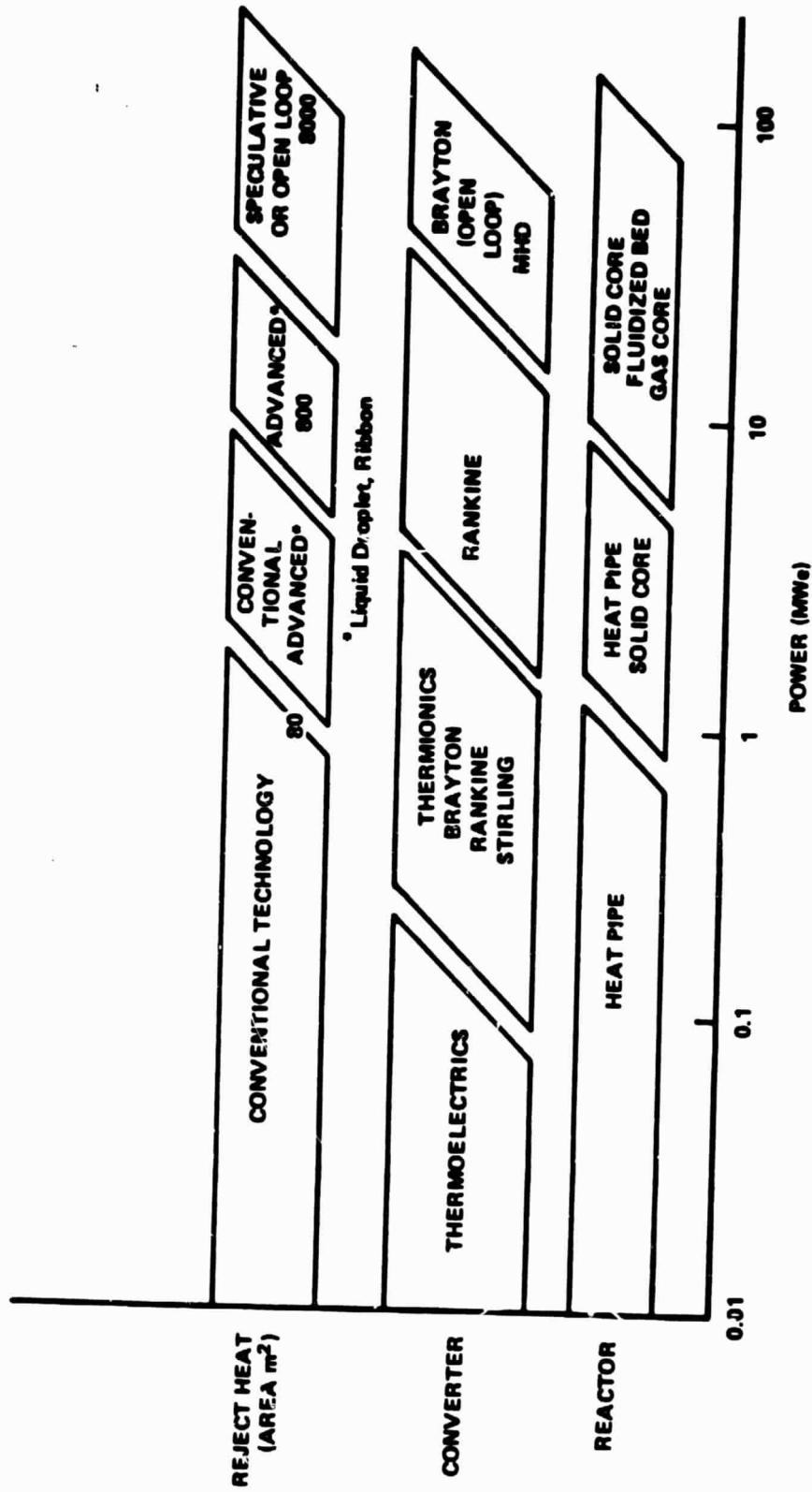
FIGURE 10 Technology candidates.
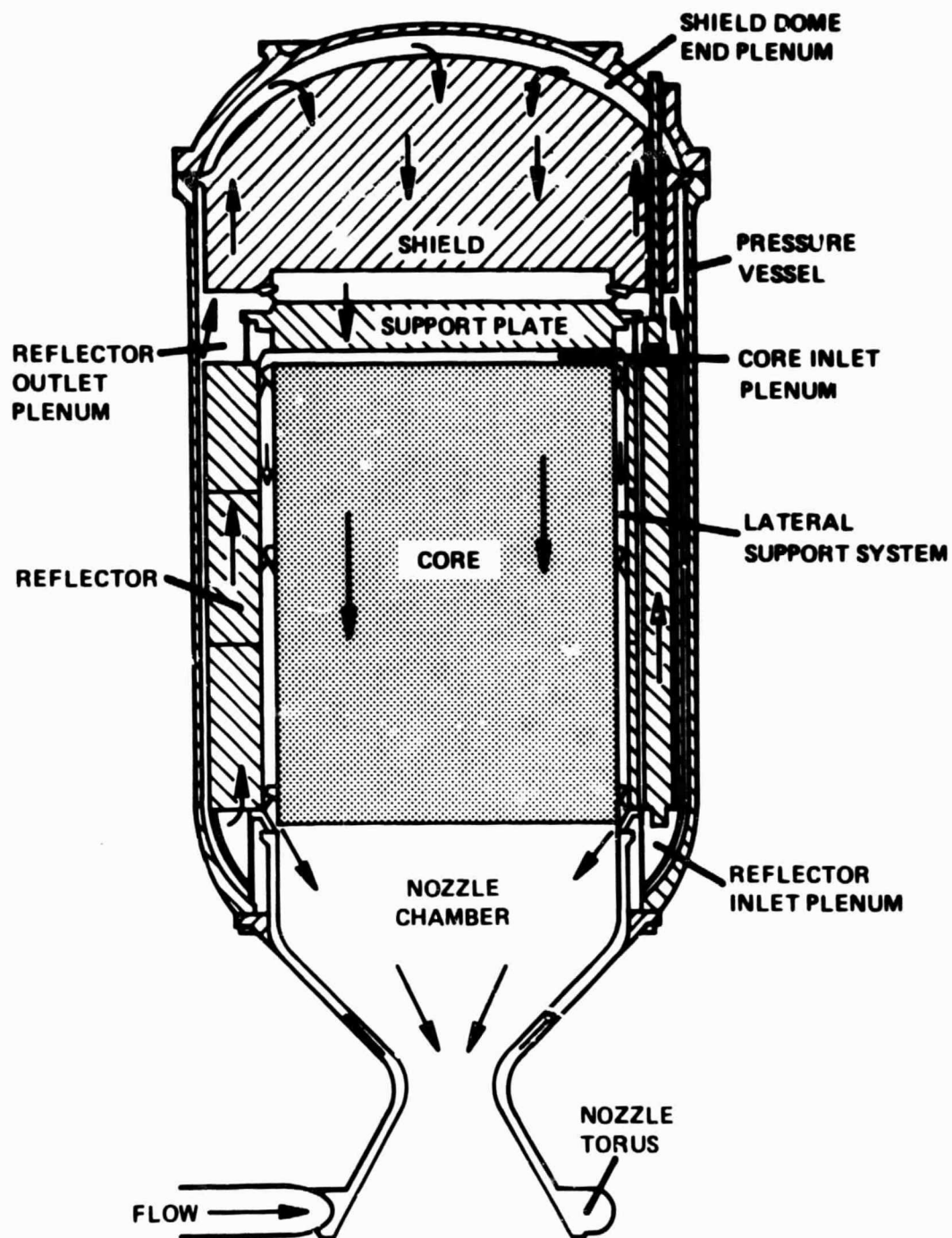
FIGURE 11 Cutaway view of Rover reactor and fuel element.

50
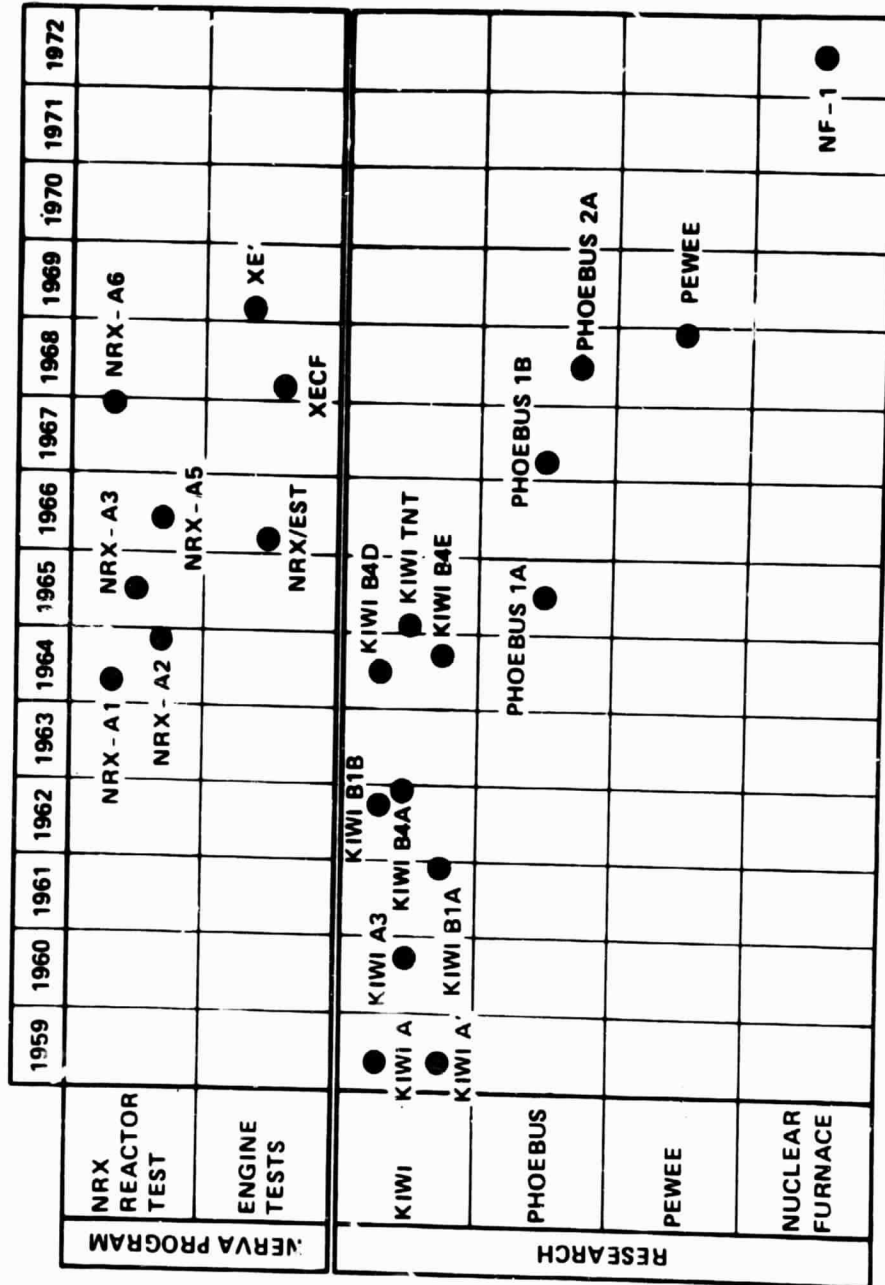
FIGURE 12   Rover technology development: major tests.
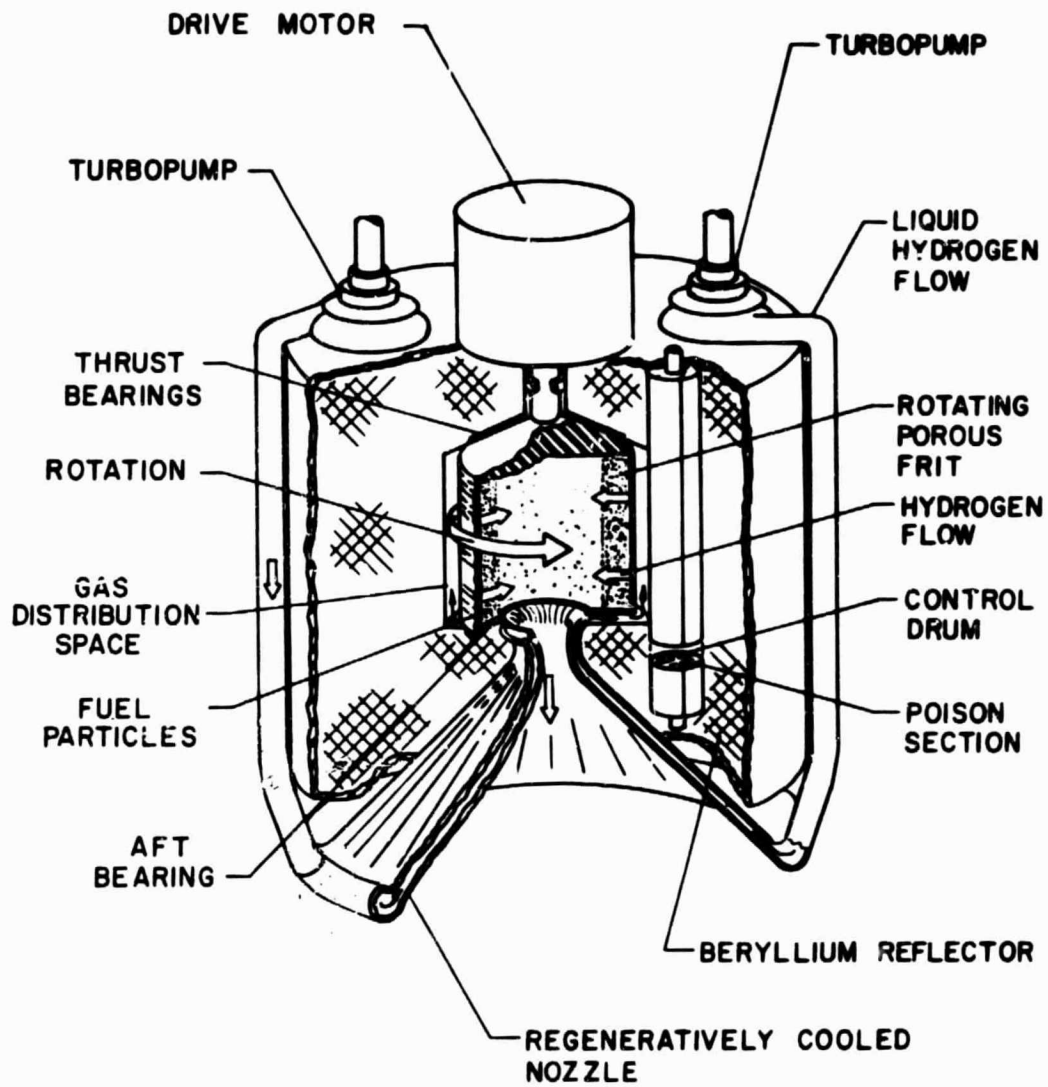
FIGURE 13 Rotating fluidized-bed rocket engine.
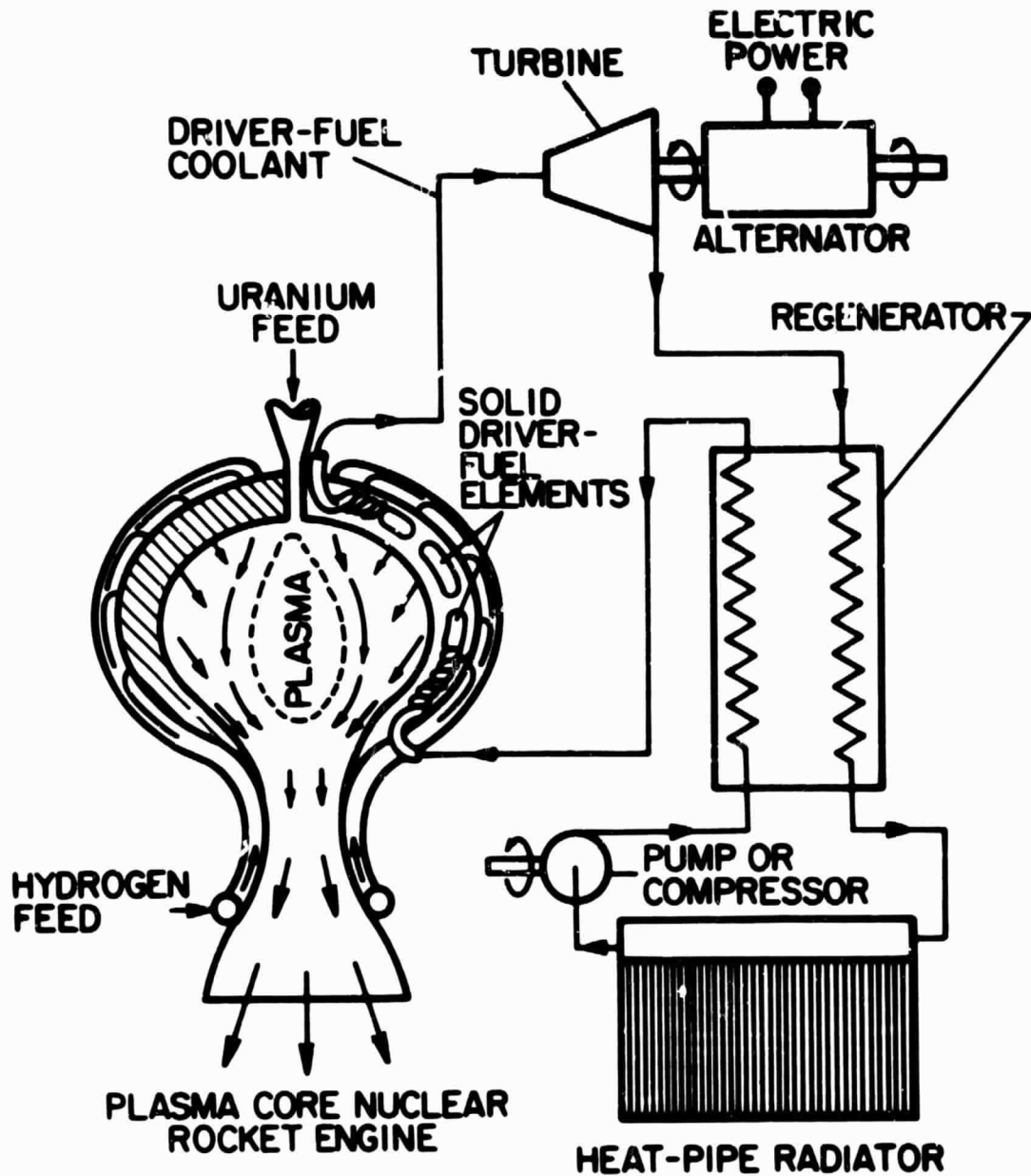
FIGURE 14 Nuclear fission plasma core dual-mode system concept.
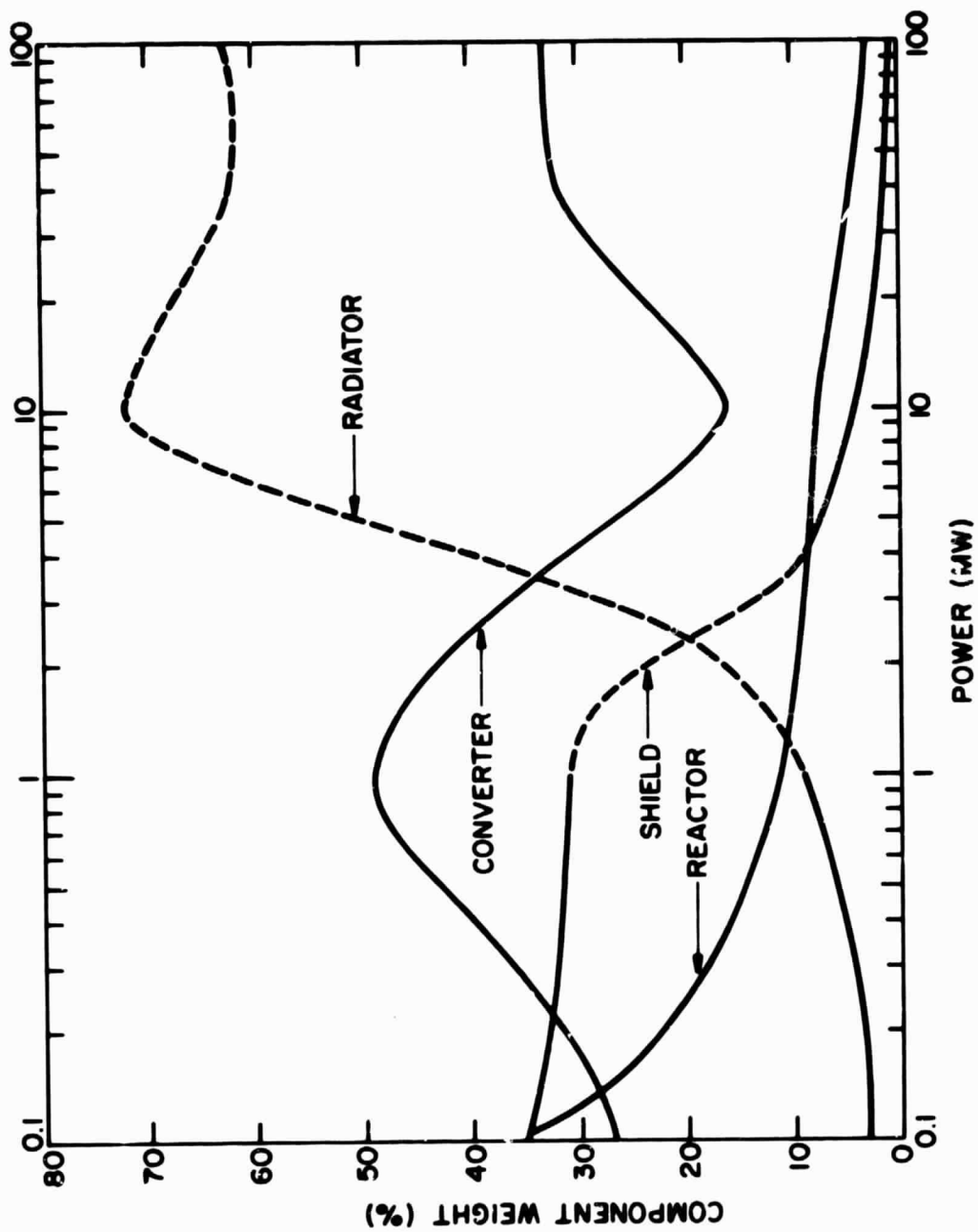
FIGURE 15 Distribution of power plant weight (percent).

SOME HIGH-TEMPERATURE REACTOR TECHNOLOGIES

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INTRODUCTION

High-temperature nuclear reactor technology has been under development
for 30 years. In the 1960s, much effort directly supported space
power and propulsion systems, but during the last 10 years, these
efforts were deferred when emphasis was placed on developing the space
transportation system. The application of nuclear power in space is
currently being reexamined to satisfy apparent needs for military
surveillance, communication, electronic countermeasures, propulsion,
and offensive/defensive directed-energy weapons. Compact,
high-performance power systems will be required to meet military
survivability criteria of hardening and maneuverability in hostile
environments. For many of these applications, nuclear power may well
be the best alternative (Buden et al., 1979; Layton et al., 1982).

Over the past 25 years, GA Technologies, Inc. (GA), formerly the
General Atomic Company, has continued to develop the high-temperature
reactor technologies that could apply to a nuclear space power system
(General Atomics, 1973a; Simon, 1982). Table 1 shows the development
programs that have contributed to these technologies: (1) metal-clad
fuel, (2) coated-particle fuel, and (3) the thermionic reactor. The
development status of these systems extends from one extreme, where
the performance and lifetime feasibility have been fully developed, to
the other extreme, where much additional basic research is needed.

The first reactor technology uses a fast reactor core consisting of
superalloy- or refractory-metal-clad fuel rods that are cooled by the
gas working fluid of a Brayton cycle. Waste heat is rejected by a
heat pipe radiator. This reactor has relatively lower potential
specific power performance than the other systems. However, both the
reactor fuel and the turbomachinery are fully developed and
demonstrated so that a prototype ground demonstration reactor could be
designed without any additional research and development. A flight
qualification reactor could be available within 6 years.

The second reactor technology uses high-temperature, coated fuel
particles for flexibility in reactor design and in selecting fuel and
coating material. The particles can be designed to completely

TABLE 1 GA High-Temperature Reactor Development Programs

Program	Power
Metal-clad fuel	
Maritime Gas-Cooled Reactor (MGCR)	70 MW(t)
Experimental Beryllium Oxide Reactor (EBOR)	10 MW(t)
Gas-Cooled Fast Reactor (GCFR)	
Demonstration plant	300 MW(e)
Commercial plant	1,200 MW(e)
Coated-particle fuel	
Peach Bottom	40 MW(e)
Fort St. Vrain	300 MW(e)
High-Temperature Gas-Cooled Reactor (HTGR)	250-1,500 MW(e)
UHTREX coated-particle-fuel production	--
NERVA coated-particle-fuel production	--
Thermionic reactor	
In-core thermionic space power reactors	10 kW(e) to 1 MW(e)
TRIGA reactor	Up to 13 MW(t)

restrain fuel swelling and retain fission products. They can be
assembled into many configurations to achieve the reactor design
objectives. Four configurations are of specific interest: (1) an
alternate core design for use in the SP-100 system, (this system is
the reference nuclear space power concept being developed by Los
Alamos National Laboratory--see the section on the SP-100 alternative
reactor concept for a further description), (2) a reactor design
similar to SP-100 but using all-carbon components and carbon heat
pipes to significantly improve temperature, (3) a core composed of
gas-cooled fuel assemblies to heat a Brayton cycle, and (4) a
gas-cooled reactor using particle fuel in a fixed or rotating bed to

provide the maximum attainable coolant temperature for use with advanced power conversion systems.

The third reactor technology uses in-core thermionic cells, based on the highly successful U.S. development program terminated in 1973 (General Atomics, 1973a). This concept was adopted by the Soviet Union for its TOPAZ space power reactor. It offers potentially very high system performance in terms of specific weight and specific volume. thermionic fuel cells have a demonstrated in-core lifetime of more than 1 year and could have extended lifetimes with further development.

Each high-temperature reactor technology is a viable candidate system for the U.S. space nuclear power program and, along with other candidate technologies, should be carefully evaluated to best meet all the criteria, objectives, and operational requirements for future U.S. space missions.

GAS-COOLED, METAL-CLAD REACTOR TECHNOLOGY

Program Background

The gas-cooled, metal-clad reactor technology of interest for nuclear space power application began in the 1950s. Gas is inherently attractive as a reactor coolant, because it is a single-phase fluid that can be operated to high temperatures at relatively low pressures. This advantage is offset by the higher pumping power required and the lower heat transfer performance of a gas relative to a liquid.

A gas-cooled nuclear reactor allows direct coupling of the reactor to a Brayton power conversion cycle. This results in a single-loop power conversion system coupled to a heat pipe radiator for redundant heat rejection. This system has a number of unique advantages, which enhance the performance and reliability of a space power system: (1) elimination of corrosion, erosion, and mass transfer; (2) single working fluid and bearing lubricant; (3) single-phase fluid, eliminating low-gravity condensing problems; and (4) simple start and restart.

The fuel element technology basis for a nuclear, Brayton-cycle space power system is provided by the following: (1) the Army Gas Cooled Reactor Systems Program (AGCRSP), (2) the Experimental Beryllium Oxide Reactor (EBOR), (3) the Gas Cooled Fast Reactor (GCFR) program, and (4) the Advanced Gas Reactor (AGR) program in the United Kingdom. The Army program included a Gas Cooled Reactor Experiment (GCRE) and the ML-1, a mobile, low-power (300 kW(e)), direct-coupled, Brayton-cycle, nuclear plant intended for military field use. This program was terminated in 1965 following the successful operation of both reactor systems. A program final report (Aerojet, 1966) includes a comprehensive program bibliography.

From 1958 through 1966, GA conducted a design and development program on a helium-cooled 10-MW(t) EBOR. These studies included successful irradiation tests of the Hastelloy X-clad, BeO_2 fuel rods. A reactor facility was built at the National Reactor Testing Station in Idaho, and a core load of fuel elements was fabricated and delivered to the reactor site. However, the project was terminated just before the reactor could start up.

From 1960 through 1979, GA and three national laboratories conducted a design and development program for a 300-MW(e) GCFR demonstration plant. The program also included cooperative agreements with Swiss and German national laboratories. A comprehensive summary of the GCFR program was published (Nuclear Engineering Design, 1977).

The AGR program in the United Kingdom included the development, design, building, and operation of 600-MW(e) nuclear power stations. This highly successful program has resulted in the construction of 10 power reactors, of which six are in commercial power operation (Nuclear Energy, 1982).

The Garrett Corporation and the National Aeronautics and Space Administration (NASA) conducted an extensive development program in the power conversion area for Brayton-cycle components and systems that can operate over a range of powers of interest in nuclear space power. Of particular interest is the successful completion at NASA-Lewis Research Center (NASA-LRC) of a 38,000-hour system test using a radial turbine, a radial compressor, an alternator, and gas bearings (English, 1982).

The four reactor programs have all helped develop high-temperature reactor fuel technology for a gas-cooled, Brayton-cycle nuclear space power system. The Army's significant fuel development program conducted in-pile tests for 10,000 hours at the environmental conditions of a space power system. A full core of fuel assemblies was operated to provide statistical performance data. Detailed heat transfer experiments verified thermal performance under gas cooling conditions with both smooth cladding and cladding with enhanced heat transfer surfaces. The fission density and fuel burnup of the tests exceeded that anticipated for the desired 7-year lifetime of a nuclear space power system. The GCFR program greatly extended the development and testing of gas-cooled, metal-clad fuel. In-pile irradiation experience was extended to neutron energies more typical of a space reactor. The extensive testing program improved, characterized, and verified clad surface heat transfer enhancement techniques. The AGR program has made operational use of this high-temperature, gas-cooled reactor technology. More than 1 million meters of fuel rods have been manufactured and operated with enhanced heat transfer surfaces.

Reactor System

Figure 1 shows the general arrangement of the gas-cooled, metal-clad reactor. The nuclear core consists of an array of Hastelloy X-clad

TABLE 2 Reactor Characteristics, Nuclear Brayton Cycle

Fuel rods	Hastelloy-clad UO_2
Clad temperature (hot spot)	1230 K
Fuel temperature (hot spot)	1420 K
Fuel burnup	0.2% per 10,000 h

UO_2 fuel rods on a triangular pitch. The fuel rods are fixed to a grid plate at the coolant inlet and are guided by a plate at the coolant outlet. The fuel rods are spaced by wire wraps and can use either smooth cladding or cladding with enhanced heat transfer surfaces. A BeO reflector is located at the core inlet.

The core outlet end is not reflected, resulting in a more effective axial power profile. A reflector/controller, located on the radius of the core, consists of movable axial segments or rotating control drums. The reflector material is BeO, and the poison segment in the control drums is B_4C. A tungsten shield section is also located at the inlet end of the core to provide both gamma radiation attenuation and elastic scattering of neutrons. The main lithium hydride shield is located between the reactor and other system components.

Table 2 lists the principal reactor characteristics. The coolant gas, a helium-xenon mixture, provides an optimum trade-off between heat transfer and turbomachinery design. The reactor performance is limited by the allowable temperature in the cladding, conservatively set at 1230 K, resulting in a minimum system weight. This low rejection temperature results in a relatively high specific weight compared with that potentially available from some of the more developmental technologies. However, the performance of this system may be attractive for early space power applications.

Brayton Cycle

The reactor is directly coupled to a Brayton cycle, which consists of a turbine, compressor, recuperator, generator, and radiator, and which uses gas as a working fluid. Table 3 lists the overall system performance parameters for a 100-kW(e) nuclear Brayton cycle. Figure 2 shows a typical set of components and cycle state points for a system using a parallel set of turbomachinery.

The working fluid leaves the reactor and flows through a single-stage radial turbine, then through a recuperator to a heat exchanger, which transfers waste heat to the heat pipe radiator. The working fluid is then compressed by a single-stage radial compressor and is preheated by the recuperator prior to reentering the reactor.

TABLE 3 System Performance, Nuclear Brayton Cycle

Power level	100 kW(e)
Specific weight	~30 kg/kW(e)
Net system efficiency	24%
Maximum system temperature	1090 K
Minimum system temperature	430 K
Working fluid	Helium-xenon (O_2 trace)

The electrical generator is mounted on a shaft between the turbine and compressor wheels.

Figure 3 shows typical Brayton-cycle components. Dynamic gas bearings, using the working fluid as a lubricant, provide axial and radial support to the combined rotating unit. Similar foil-type gas bearings have been in service on commercial DC-10 aircraft for more than 50 million hours with a mean time between failure of 250,000 hours (Garrett Corporation, 1979). Gas injection provides start and restart. The alternator load controls turbine speed, and a radiator bypass controls the compressor inlet temperature.

An attractive feature of this system is that the major components (reactor core, turbine, compressor, and recuperator) can be operated over a wide range of power by varying the pressure roughly in proportion to the desired operating power. However, the electric generator and heat rejection system must be designed for the specific power loads. Thus the design and flight qualification of the initial nuclear power system components are directly applicable for systems at other power levels.

Summary

Gas-cooled, metal-clad nuclear reactor fuel technology has been developed and demonstrated at conditions that meet, and in most cases exceed, the requirements for a nuclear space power system. Brayton-cycle turbomachinery, bearings, and heat exchanger components are at the same high state of development. This technology can form the basis for designing a prototype space power plant for ground operation and for fabricating a nuclear power system for flight qualification. An industrial infrastructure is in place and available to undertake this task quickly. The facilities necessary for production and engineering testing are also available. A nuclear space power system could initiate flight qualification within 6 years.

This system could be designed and fabricated for a fraction of the cost of a more developmental system. Early flight testing would provide launching, operational, and spacecraft integration

experience. This nuclear space power system could also meet interim space power needs until a more advanced system is developed in perhaps 10-15 years.

The overall specific weight of this system is expected to be somewhat higher than that of a more advanced system because of the lower heat rejection temperature. However, the power system is still compatible with the operational criteria of the space transportation system.

COATED, SPHERICAL FUEL PARTICLE REACTOR TECHNOLOGY

Program Background

In 1945-1947, work on coated particle fuels began with the Daniels' Pile at Oak Ridge National Laboratory. The proposed reactor fuel consisted of UO_2 particles dispersed in a graphite matrix. In the late 1950s, GA initiated the High-Temperature Gas-Cooled Reactor (HTGR) program. Pyrocarbon-coated fuel particles of UC_2 and $Th-UC_2$ were developed to prevent hydrolysis of the carbide fuel particles upon exposure to air. Both the initial GA and the Dragon (U.K.) reference fuel designs consisted of vented fuel in impermeable-graphite sleeves. When fabrication of low-permeability graphite turned out to be difficult and expensive, GA and Dragon developed coated particles to retain fission products (Goeddell, 1967; Simnad, 1971, 1982).

The Peach Bottom HTGR, a 40-MW(e) prototype reactor power plant project, with fuel based on the particle technology, was initiated in 1957 and decommissioned in 1974. The Peach Bottom fuel element consisted of a sleeve of low-permeability graphite, containing a fuel-bearing middle section, top and bottom reflector sections, and an internal fission product trap. Annular grooved fuel compacts, consisting of single-layer, pyrocarbon-coated fuel particles dispersed in a graphite matrix, were stacked on a cylindrical graphite spine. The single-layer, pyrocarbon coatings merely protected the carbide fuel particles from reaction with the atmosphere. A continuous flow of helium purge gas removed the fission products to a trapping system. The first core in Peach Bottom was replaced in June 1970, after achieving a burnup of 30,000 MWd/t. The Core 2 fuel elements utilized coated particles that were designed to retain all fission products within the particle, and experience from 1970 to 1974 showed this fuel to be very successful.

As this fuel technology evolved, the fuel element design for the large HTGR reactors shifted to a hexagonal block with channels containing rods of coated fuel particles interspersed with channels for coolant flow (Figure 4). This fuel element was based on the need to produce a structurally sturdy and simple fuel element for use in very large reactor cores. This design has been successfully implemented and is being used with excellent results in the 330-MW(e)

Fort St. Vrain HTGR in Colorado. To date, 10^{12} fuel particles have been manufactured and used successfully in support of HTGRs. Extensive work has also been completed on higher temperature, coated-particle fuels for the unique applications of HTGRs in direct Brayton-cycle conversion and for process heat (Gulden and Nickel, 1977; Gulden and Watson, 1982).

The use of coated fuel particles allows the core to operate at high temperature to very high burnups (more than 70 percent of the fissile fuel) with essentially complete retention of fission products and excellent dimensional stability. Two types of coatings have received the most attention for the HTGRs (1) BISO, which consists of two carbon coatings, i.e., a low-density porous buffer inner coating and an isotropic pyrocarbon outer coating, and (2) TRISO, which consists of four coatings, i.e., an inner porous buffer carbon, isotropic pyrocarbon, silicon carbide (SiC), and an outer isotropic pyrocarbon.

The inner buffer layer of low-density pyrocarbon protects the outer layer from fission recoil damage and provides void space to accommodate the fission gases, fuel swelling, and coating contraction. The SiC layer in the TRISO coatings decreases the release of certain metallic fission products that migrate readily through pyrocarbon (e.g., barium, strontium, cesium).

Coated-particle fuel was used in the Rover and NERVA nuclear rocket programs (Taub, 1975). This fuel has a single coating of pyrolytic carbon, and the particles were bonded in a graphite matrix in the form of hexagonal rods with integral coolant holes (see Figure 5). These particles had a relatively thin coating designed to operate to relatively low burnups. The fuel particle packing fraction was low, allowing sufficient matrix material to provide the needed structural strength for the fuel element when subjected to high-velocity coolant flows and core vibration. The fuel elements were cooled with hydrogen and achieved an outlet temperature of 2500 K with a design lifetime of several hours. Los Alamos National Laboratory (LANL) also developed and tested composite fuel elements containing graphite and approximately 35 vol% (U,Zr)C carbide and fuel elements consisting only of (U,Zr)C carbide for the Rover program (Lyon, 1973).

The use of ZrC coatings as a barrier to fuel kernel migration and fission product diffusion at very high temperatures (over 1700 K) has also been studied with encouraging results (Gulden and Watson, 1982). Good performance for extended lifetimes appears feasible.

The W-UO_2 nuclear fuel elements, developed in the 1960s by Argonne National Laboratory and NASA-LRC as backup nuclear rocket fuel concepts, consisted of UO_2 particles dispersed in a tungsten matrix (Holden, 1967; Rom, 1968; Speidel, 1968). This fuel was fabricated in the form of hexagonal elements, containing hexagonal honeycomb coolant channels, by hot isostatic compaction of tungsten-coated spherical UO_2 particles. Molybdenum dummy rods were positioned in the matrix for coolant channels and dissolved out after hot isostatic compaction.

This paper describes four high-temperature reactor concepts that utilize coated particles: (1) an SP-100 alternative reactor concept,

(2) an SP-100 advanced reactor concept, (3) a gas-cooled, graphite-clad reactor concept, and (4) a gas-cooled reactor concept using particle fuel in a fixed or rotating bed.

SP-100 Alternative Reactor Concept

The SP-100 nuclear space power reactor is being developed by LANL under the sponsorship of the Department of Energy (DOE) (Boudreau, 1982; Buden and Stocky, 1982). The system produces 100 kW(e) and consists of a reactor core cooled by heat pipes that thermally radiate to a thermoelectric power conversion system mounted directly on the radiator surface. Each heat pipe forms a fuel element module in the core region. The heat pipe has 0.5-mm-thick integral fins interspaced by 2.0-mm-thick UO_2 fuel tiles. The fuel modules are assembled together in a cylindrical core configuration surrounded by a structural vessel that is vented directly to space. The core has a radial reflector that incorporates control drums, axial reflectors, and a central cavity that houses a neutron-absorbing plug (for safety during the launch and preorbital flight to prevent flooded criticality). The plug is removed for reactor operation.

The reactor fuel operates at a maximum nominal temperature of about 1750 K. When a heat pipe fails, the fuel temperature will approach 2000 K. Concerns have been expressed regarding the swelling of the fuel over its 7-year operation lifetime and the mechanical interaction with the finned heat pipe. The fuel has been designed to mitigate the effects of fuel structrual interaction, and an in-pile test program has been initiated to evaluate fuel performance under irradiation conditions. Concerns also relate to fuel migration, fission product mobility, and interaction between the fuel and the lithium working fluid of a failed heat pipe.

Figure 6 shows the alternative fuel concept being developed at GA under DOE sponsorship, which retains the general configuration and the heat pipe concept of the SP-100 reactor and which answers these concerns. This concept replaces the fuel tiles and heat pipe fins of the SP-100 core with coated, spherical fuel particles that may be bonded together in a matrix surrounding the heat pipe. The overall dimensions of the alternative heat pipe fuel module are similar to those of the reference SP-100 design.

The coated-particle fuel design is intended to (1) restrain fuel swelling, (2) prevent fuel migration, (3) contain fission products, and (4) prevent fuel-coolant interaction. When combined into a matrix to form a fuel compact, the concept will (1) increase thermal conductivity, (2) reduce fuel temperature, and (3) provide fuel structural support. The use of coated-particle fuel allows flexibility in choosing materials for fuel, fuel coating, and matrix.

SP-100 Advanced Reactor Concept

The heat pipe temperature of the reference SP-100 design is limited by the material capabilities of the fuel and the refractory metal structural material (molybdenum-rhenium) used for the heat pipe and fins. Molybdenum-rhenium is a high-density material with a relatively high parasitic neutron capture. The reference SP-100 reactor design concept cannot achieve temperatures that can be effectively used with out-of-core thermionic power conversion or thermophotovoltaic power conversion. (The thermophotovoltaics converter consists of a high-temperature emitter, such as tungsten, that thermally radiates to a photovoltaic collector. The photovoltaic material operates at a low temperature (about 600 K) and is backed with a reflector to return the photons not absorbed back to the emitter. The conservation of energy is the basis for the relatively high efficiency of the thermophotovoltaic device.) Thermophotovoltaics is of particular interest, since it has the potential of high efficiency (more than 25 percent).

An advanced SP-100 concept using a graphite heat pipe and coated-particle fuel could increase the reactor and heat pipe performance by at least 250 K. This would result in heat pipe temperatures of 1750 K, within the range of interest for high-temperature power conversion systems.

The advanced design concept, similar in configuration to the proposed alternative SP-100 design shown in Figure 7, is contingent on the feasibility of a graphite heat pipe. While more than 30 years' experience exists on the use of graphite in nuclear reactors, heat pipes have not been made from carbon materials, and a number of fundamental development issues need to be resolved. Graphite has excellent high-temperature strength. For operation out of the core, where radiation levels are lower, a carbon-carbon composite could provide the exceptional strength and stiffness needed for the heat pipes to sustain dynamic loading during launching. Carbon materials tend to be porous; however, gaseous- and liquid-phase impregnation techniques have been developed to reduce permeability to acceptably low levels. Carbon heat pipes would also require a working fluid that is chemically compatible with carbon and that meets the thermal and flow criteria for the heat pipe application. A number of promising candidate working fluid materials are available.

The nuclear fuel concept uses coated, spherical-shaped fuel particles contained in a matrix material that bonds them to the heat pipe. The fuel would be uranium carbide coated with a thin buffer layer, then a zirconium carbide layer. The coated particles would be made in two sizes, with a diameter ratio of 7 to 10, and would be vibratory compacted into a high-density compact. A hydrocarbon binder fluid (e.g., pitch) with a powder graphite suspension would be injected into the compact and graphitized to provide the matrix material. The compact could be formed integrally with the graphite heat pipe or subsequently bonded to the heat pipes.

The use of carbide fuel will increase fuel loading and thermal conductivity more than the use of the reference oxide fuel. The coating and matrix material also increases the overall conductivity, resulting in lower fuel compact temperatures. The working fluid, which is compatible with the heat pipe, will also be compatible with the fuel coating and matrix materials.

Gas-Cooled, Graphite-Clad Reactor Concept

The performance of a nuclear space power system that utilizes a direct Brayton cycle can be significantly improved by increasing turbine inlet temperature. A major effort was initiated in the United States to develop a high-temperature, gas-turbine-drive system for automotive applications that are in about the same power range as the space power application. The turbine is a single-stage, radial, in-flow wheel. The inlet temperature is limited by the turbine material and the inlet duct material. The use of ceramic materials for ducts and turbine wheel can accommodate turbine inlet temperatures up to 1500 K, increasing net system efficiency and reducing system specific weight to about 20 kg/kW(e).

A graphite-clad, coated-particle fuel (see Figure 8) could attain a 1500 K outlet temperature. This fuel concept is similar to that used in the HTGR. The fuel is a spherical particle that is coated to restrain swelling and retain fission products. The fuel particles are in two sizes to provide maximum packing fraction and bonded together in a matrix material to form a fuel compact. The fuel compacts are assembled in a high-strength graphite clad, which provides structural support for the fuel. The fuel rods are gas cooled by an inert gas mixture that has a molecular weight selected to optimize system specific weight.

The fuel rods are assembled into a cylindrical core configuration. They are attached at the coolant inlet end by a support plate and laterally supported at the outlet by a guide plate. The core is reflected by BeO, and cylindrical control drums are located around the core circumference. The reactor operates in a fast-neutron spectrum because of its small size and high neutron leakage rate. The high temperatures in the core will minimize neutron irradiation damage in the graphite cladding and matrix by annealing the damage.

Gas-Cooled Particle Bed Reactors

A reactor can attain by far the highest coolant temperatures by using coated-particle fuel that is directly gas cooled in either a fixed or rotating bed. The very high heat transfer coefficients result in low fuel surface-to-gas temperatures. Also, high surface-to-volume ratios of the fuel particles result in low heat fluxes. Since the particles are small, the fuel temperature rises very little, resulting in a

coolant temperature very close to the fuel particle center temperature that limits the system. As a consequence, the coolant outlet temperature can exceed 2500 K. The fuel is in the form of spherical particles coated in a configuration similar to HTGR fuel, including an SiC or ZrC coating protected by an outer pyrolytic carbon coating.

The reactor core can operate at very high power. The reactor cores are constructed in the form of fixed or rotating beds. (These are under development at Brookhaven National Laboratory (Powell, 1982).)

Summary

The basic coated fuel particle concept represents a well-established fuel technology that can be used in a wide variety of reactor design concepts. The coated particles restrain fuel swelling, contain fission products, prevent fuel migration, and prevent fuel-coolant interaction. A fuel compact composed of coated fuel particles incorporated into a matrix material increases conductivity, reduces fuel temperatures, and provides fuel structural support. Directly cooled coated fuel particles can achieve very high working fluid temperatures.

IN-CORE THERMIONIC REACTOR TECHNOLOGY

Program Background

Programs to develop in-core thermionic power systems for nuclear space power were initiated in the early 1960s in the United States, and parallel programs were carried out in France and Germany with cooperative agreements for information exchange. Although these programs achieved considerable success, they were terminated in the United States in 1973 (General Atomic, 1973a,b). However, the Russian in-core thermionic program has continued and culminated in the development of the TOPAZ thermionic space reactor, which is in operational use in the USSR space program (Kuznetzov et al., 1974). International conferences were conducted on thermionic electrical power generation under the joint sponsorship of the International Atomic Energy Agency (IAEA) and the European Nuclear Energy Agency (ENEA). The proceedings of these conferences comprehensively summarize the state of development of both in-core and out-of-core thermionic technology (IAEA, 1972).

The U.S. program for thermionic reactor development was supported by the Atomic Energy Commission and the Energy Research and Development Administration (AEC/ERDA). Thermionic fuel material and most out-of-reactor converter development and testing was supported by NASA and the Department of Defense. Systems studies were supported by both AEC/ERDA and NASA. General Electric and GA conducted parallel programs to develop nuclear-fueled thermionic converters as a

TABLE 4 Typical Thermionic Converter Operating Conditions

Cathode	Heated tungsten
Operating temperature	2000 K
Anode	Niobium or molybdenum
Operating temperature	1000 K
Cesium vapor pressure	10^{-1} MPa
Cathode-to-anode gap	~0.2 mm
Current density	~10 A/cm^2
Net operating voltage	~0.5 V
Electrical power density	~5 W/cm^2
Net converter efficiency	10%
Heat flux at collector	~45 W/cm^2

technical competition until 1970, when GA was selected to develop the in-core thermionic reactor for space power applications. This program included the fabrication and in-core testing of single-cell and multicell thermionic fuel elements, a thermionic reactor critical experiment, the design and planning for a thermionic reactor experiment, and the design of thermionic reactors and power systems for space and remote terrestrial applications. A TRIGA Mark III testing reactor was built for life testing of up to 15 in-core thermionic fuel elements (TFEs) simultaneously. The fabrication and testing programs were highly successful, with in-core converter lifetimes steadily increasing to more than 10,000 hours of power operation. In January 1973, ERDA canceled the program along with all other nuclear space power and propulsion programs that did not have an approved mission.

Thermionic Power Conversion Principle

Thermionic emission is a well-known phenomenon that provides a basis for the electron tube. Thermionic emission can also convert heat directly into electricity (Chang, 1963; Kettani, 1970). Figure 9 shows the basic components of a thermionic converter, and Table 4 gives typical operating conditions. A heated cathode emits electrons that travel across a narrow gap to a relatively cool anode. The cathode and anode are connected through an external load impedance that completes the electrical circuit and establishes a potential difference. To enhance the performance of the thermionic converter, the space charge in the gap between the cathode and anode is suppressed by an ionized metallic vapor, usually cesium. Heat is transferred from the cathode to the anode by electron emission, thermal radiation, and cesium vapor conduction. The maximum potential

available to the load is the difference of the cathode and anode work function, as modified by the adsorbed cesium vapor.

Thermionic Cell Design

Thermionic converters can generate high power densities of the same magnitude as nuclear reactor fuel, and the required cathode temperatures are consistent with the temperature capability of several fuel forms, including uranium dioxide and uranium carbide. The basic thermionic power conversion process is not degraded by nuclear radiation. Fueled cylindrical thermionic cells can operate in the core of a nuclear reactor.

Typically, six cells are connected in series electrically and assembled into a thermionic fuel element (TFE). In this design, the fuel cladding material is also the cathode of the thermionic converter. Heat generated in the nuclear fuel is conducted directly to the cathode, and electrical power is generated in the space between the cathode and anode. An electric insulator, with relatively good thermal conductivity, and metal sheath separate the anode from the coolant.

This in-core thermionic reactor concept has several inherent advantages over other reactor concepts:

1. Since the reactor thermal power is applied directly to the power conversion components, energy need not be transported in a high-temperature pumped loop or heat pipe system for reasonable power conversion efficiency.

2. Since the anode temperature is slightly higher than the coolant temperature, a relatively small radiator can reject waste heat, giving the reactor a high specific power.

3. The TFEs can provide redundancy to safeguard system performance against individual TFE failure.

4. The failure of a TFE by shorting or an open circuit will not propagate outside the TFE.

5. Since the system is composed of a large number of unit cells, the principal development can be accomplished on a unit cell level. Subsequently, the unit cells can be assembled into an operating system with the desired power, thus saving considerable development cost and time.

These advantages are accompanied by some inherent difficulties:

1. The TFE design requires relatively small interelectrode spacing, which limits cathode distortion resulting from fuel swelling.

2. Diffusion of fuel materials into the emitter and subsequent transfer to the collector can change electrode work functions and reduce electric performance.

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TABLE 5 Thermionic Fuel Element Materials

Emitter	
Tungsten	Inner 0.7-mm vapor deposited from WF_6
	Outer 0.3-mm vapor deposited from WCl_x
Surface	Electropolished (110) orientation
Collector	Niobium
Electric lead	Molybdenum
Outer sheath	Nb-1% Zr
Fuel material	90% UC-10% ZrC plus 4 wt% tungsten
Range of carbon/uranium ratio	$1.01 \leq C/U \leq 1.03$
Fuel body theoretical density	77% av
Maximum emitter cavity loading	69% of theoretical density
Insulator-seal	Brazed Nb-Al_2O_3 (Lucalox)-Nb
Sheath insulator (trilayer)	Gas-pressure-bonded Nb-Al_2O_3/Nb cermet (plasma sprayed)-Nb

3. Insulation and sealing materials are also subject to fast-neutron damage and thermal degradation.

Past development programs have demonstrated TFE performance in excess of 10,000 hours. A continued development effort is necessary to determine the lifetime potential of in-core thermionics and the overall TFE reliability.

Figure 10 shows the design of a typical 50-mm-long, 30-mm-diameter thermionic cell and the alignment spring and ceramic insulator used to connect it to the next cell in a TFE for irradiation testing. The emitter is made of a two-layer tungsten cup containing fuel pellets of UO_2 or 90% UC-10% ZrC. The outer surface layer of the emitter has a (110)-oriented tungsten grain structure (approximately 5.0-eV base work function) coated on a (100)-oriented tungsten grain structure with high creep strength and stable grain structure. An insulating layer of alumina bonds a cylindrical niobium collector to an outer sheath of niobium-1% zirconium; the collector is aligned concentric with the emitter, with a 0.2-mm radial gap. An insulating ring of alumina joins the emitter and the collector together to seal the cesium vapor in the interelectrode gap without shorting the electrodes.

Thermionic Fuel Element Design

A typical TFE design (shown in Figure 11) consists of six individual cells assembled in the sheath tube with end reflectors of BeO and electrical leads. Table 5 gives the TFE fabrication materials.

The cells are connected electrically in series; the cell spacing and alignment are maintained by the electrical connection at the top of the cell and an alignment spring and ceramic insulator at the bottom of the emitter. The cathode of each cell is electron beam welded to the anode of the cell above it. Connecting holes between cells provide access for cesium vapor from a reservoir outside the reactor vessel. A passage in the electrical lead connects the reservoir to the TFE. Long-time operation of the TFE requires fission product gas to be vented from the emitter fuel cavities through passages in the sheath tube insulation and then through a passage in the electrical leads to external fission product traps.

After the six cells are welded together, the assembly is plasma sprayed with Al_2O_3 in the intercell regions and assembled into the sheath tube. A degassing and bonding process completes the assembly.

The TFEs are made in two sizes: the E series, with a 1.6-cm emitter diameter and 2.1-cm outside diameter for megawatt-class reactor applications, and the F series, with a 2.8-cm emitter diameter and 3.3-cm outside diameter for lower power levels, where reactor criticality limits require higher fuel volume fractions. Both size TFEs have a cell length of 7.4 cm, with a fuel length in each cell of 5.1 cm.

In developing the thermionic cells and TFEs, GA developed a great deal of material and fabrication technology. The following is a summary of a few of the most significant technological developments:

o Stable UO_2 plus UC-ZrC fuel clad for more than 10,000 hours

o Chemically vapor-deposited (CVD) tungsten emitters with stable grain structure at 2000 K

o Oriented (110) CVD tungsten emitter surface coatings for improved electrical performance

o Diffusion bonds of CVD tungsten to tantalum

o Al_2O_3-to-niobium insulator seal for 1200°C service

o Gas-pressure-bonded Al_2O_3-Nb trilayer sheath insulator

o Refractory metal outgassing and welding techniques for cell and TFE assembly

o Diffusion bonds between niobium inner and outer sheaths.

In-Core Testing

Thermionic devices and fuel-clad capsules were tested in the thermionic test reactor (TITR) at GA. The in-core tests were intended to demonstrate improved lifetimes by upgraded components and to determine how a combination of fuel and fission product diffusion and the nuclear environment affected performance. The tests included device start-up, performance mapping, periodic logging of data, diagnostic measurements, and neutron radiographs; analysis and interpretation of the data obtained followed.

Thirty-seven thermionic devices underwent in-pile life tests between 1962 and 1973 (see Table 6). In-core converter lifetimes were considerably improved, with the longest tests increasing from 1,000 to 2,000 hours in 1965 to more than 12,000 hours by 1972. About three or four devices were tested per year until mid-1971, when a new core installed in the TITR (coincident with the start-up of the first full-length TFE, 6F1) allowed 10-15 simultaneous tests per year. This increased number of tests increased the information available to the development program and shortened the time to assess improvements.

Figure 12 gives the maximum lifetimes and representative performance levels of the TFEs. The longest in-core TFE test was by TFE 2E1, a two-cell element that operated for 12,535 hours at 1900 K maximum. The second longest test was by TFE 2E2, which was still in operating condition when the program closed.

Figure 13 shows the operating performance of TFE 2E2 over the full operating lifetime. Its operation included thermal cycling due to 10 reactor shutdowns and 11 trips. Its performance was stable within ± 10 percent throughout the test. Radiographs taken about every 2,000 hours over the operating lifetime of the element showed no evidence of fuel-cladding reactions.

As shown in Figure 11, TFE 2E2 was just one of six TFEs still in operating condition when the program closed in January 1973. These TFEs contained improvements in fuel configuration and composition that were expected to improve the dimensional and chemical stability of fueled emitters and increase TFE lifetime. In January 1973, testing of an electrically heated life-test cell was also terminated. This cell had operated in the laboratory for more than 5 years and demonstrated the long-lifetime potential of thermionic converters.

Thermionic Core Design

Figure 14 shows the general arrangement of a typical thermionic reactor core. The fuel elements are arranged on a triangular pitch and supported between two plates. They are welded to the upper plate through which the electrodes extend and are guided by the grid plate at the other end. The fuel elements incorporate an axial reflector. The core is surrounded by a radial BeO reflector with integral control drums. The control material is a B_4C segment on the control drums. A liquid-metal coolant enters the reactor vessel and flows up through an annular passage around the core. When the coolant reaches the plenum at the top of the core, it flows radially inward, then back through the passages between the fuel elements to remove the waste heat from the core. The coolant then flows through a tungsten gamma shield to the coolant outlet nozzle.

The core is composed of one or two types of fuel assemblies, depending on the system power capability. For higher power, the core is composed entirely of TFEs. A minimum of about 162 TFEs are required to achieve nuclear criticality; this provides 120 kW(e).

TABLE 6 Summary of GA Thermionic In-Pile Life Tests

| | Experimental Cells | Prototype Cells | | Thermionic Fuel Elements | | | |
	Mark VI	Mark VIIA	Mark VIIB	E Series (Two Cell)	F Series (One Cell)	(Two Cell)	Six Cell
Number of tests	15	3	4	3	4	3	5
Total test (h)	45,449	4,604	15,915	27,280	22,152	16,551	23,972
Average test (h)	3,030	1,535	3,979	9,093	5,538	5,517	4,794
Power density (W/cm²)	6.3	5.2	3.7	4.5	3.8	4.0	2.6
Longest test (h)	9,754	2,006	7,881	12,535	8,560	8,021	8,062
Power density (W/cm²)	6.0	6.2	3.4	4.0	4.0	3.4	3.0

Power can be increased into the megawatt range by increasing the core
size and the number of TFEs. For power below 120 kW(e), fewer TFEs
are required. Critical reactor configurations are attained by using
uranium-zirconium hydride (U-ZrH) driver fuel elements 1 cm in
diameter, similar to SNAP or TRIGA reactor fuel elements. For power
from about 40 to 100 kW(e), individual TFEs are each surrounded by
U-ZrH fuel elements in a critical reactor configuration (Homeyer,
1969). This arrangement is similar to actual test configurations of
the thermionic test reactor used for in-core TFE irradiations. For
lower power, additional driver fuel is arranged around the critical
core, which contains TFEs and driver elements. Figure 15 shows
typical reactor configurations.

Thermionic Reactor System Design

Figure 16 shows a typical arrangement of a thermionic power reactor
system for a manned space station (General Atomics, 1968; Gietzen et
al., 1971). Primary liquid-metal coolant is circulated through the
reactor by electromagnetic pumps. A compact heat exchanger transfers
heat from the primary to the secondary coolant. The secondary loops
distribute the heat to radiator panels, which are composed of many
heat pipes to provide a highly redundant, lightweight, and nearly
isothermal radiating surface. For the higher-power reactor systems
composed entirely of TFEs, the cooling system can operate at about
1000 K, resulting in a relatively small radiator. For lower power
systems using U-ZrH driver assemblies, the coolant temperature must be
lower (about 800 K) to be compatible with the temperature limits of
the driver fuel elements.

The radiation shields consist of lithium hydride to attenuate
neutrons and tungsten to attenuate gamma rays. For manned
applications, a relatively thin primary shield at the side and top of
the reactor limits the dose rate to 100 R/h at 50 m in any direction.
A thick, multilayer, secondary shield at the bottom of the reactor
protects occupants of the space station. The primary shield
attenuates radiation from the reactor and protects the secondary
coolant from activation. The secondary shield attenuates the gamma
radiation from the activated primary coolant in the heat exchanger and
further attenuates radiation emitted from the reactor. The size and
configuration of the shield depend on the dose criteria for the
payload and the specific spacecraft configuration.

The radiator is located within the shadow of the shield to reduce
secondary scattered radiation. Power conditioning equipment is
located between the radiator and the secondary shield and arranged to
reject waste heat by direct radiation. To increase TFE lifetime and
minimize fuel clad (emitter) distortion, the gaseous and volatile
nuclear fission products are vented from the core and collected in
external fission product traps. This maintains internal pressure in
the fueled components at about 0.1 MPa. The control drum drive units

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are also mounted outside of the reactor, either as shown in Figure 16
or behind the primary shield.

Summary

By the conclusion of the in-core thermionic reactor program in 1973,
the development tests on the thermionic fuel element design had
confirmed the basic feasibility of an in-core thermionic reactor. The
program demonstrated TFE lifetimes of greater than 11,000 hours, and
design improvements were proposed to further extend the operating
lifetime. An electrically he   ᵈ life-test cell operated for more
than 5 years. Overall, system designs had been prepared for missions
from 10 kW(e) to 1 MW(e).

The in-core thermionic reactor concept has several unique features
that make it an attractive candidate space power system:

1. Since the thermionic conversion process is directly coupled to
the nuclear fuel, energy need not be transported in a high-temperature
heat pipe system or pumped loop to achieve reasonable power conversion
efficiency.
2. Since the neat rejection system can operate at a relatively
nigh temperature, resulting in a small, lightweight, highly efficient
radiator, t e reactor system can achieve a nigh specific power.
3.   eactor can operate over a wide range of power utilizing
the same TFE module.
4. The module fuel element also can easily perform multiple
irradiation tests in a standard TRIGA research reactor.

Altnough this program was terminated almost 10 years ago, key
.members of tne original development program staff still work
togetner. Fabrication and testing facilities can be quickly
replicated, and the in-core tnermionic program can be reinitiated with
minimum delay. The in-core thermionic reactor concept has already
achieved significant developmenta¹ succ   and shows sufficient
potential to be a strong candidate nuc.   space power system.

CONCLUSION

Tnree nign-temperature reactor technologies nave been described here
that can provide a basis for one or more nuclear space power systems.
Figure 17 lists these systems and qualitatively compares their
relative merits against those of the SP-100 reference design concept
for (1) development status, (2) specific power, (3) potential
lifetime, and (4) power potential. It rates each nuclear space power
system as high, medium, or low in each category. Of course, each
rating varies widely, is very subjective, and is intended only to
provide an initial relative perspective of the systems.

The development status of these systems extends from one extreme, where the performance and lifetime feasibility have not been demonstrated, to the other extreme, where the systems have fully developed technology ready for final design and prototype production. In the midrange are development programs to establish levels of component reliability.

Figure 17 lists the reactor power systems in the approximate order of increasing system temperature. It shows expected trends in specific power (kW(e)/kg), potential lifetime (hours of operation), and potential maximum system power. Specific power should generally increase with temperature; potential lifetime should decrease with increasing temperature; and power potential should also increase with system temperature (except in the case of a reactor cooled with heat pipes, where the power density is lower and the required number of heat pipes cannot be substantially increased).

All candidate nuclear space power systems should undergo a much more quantitative evaluation to cover a broader range of significant system performance parameters.

ACKNOWLEDGMENTS

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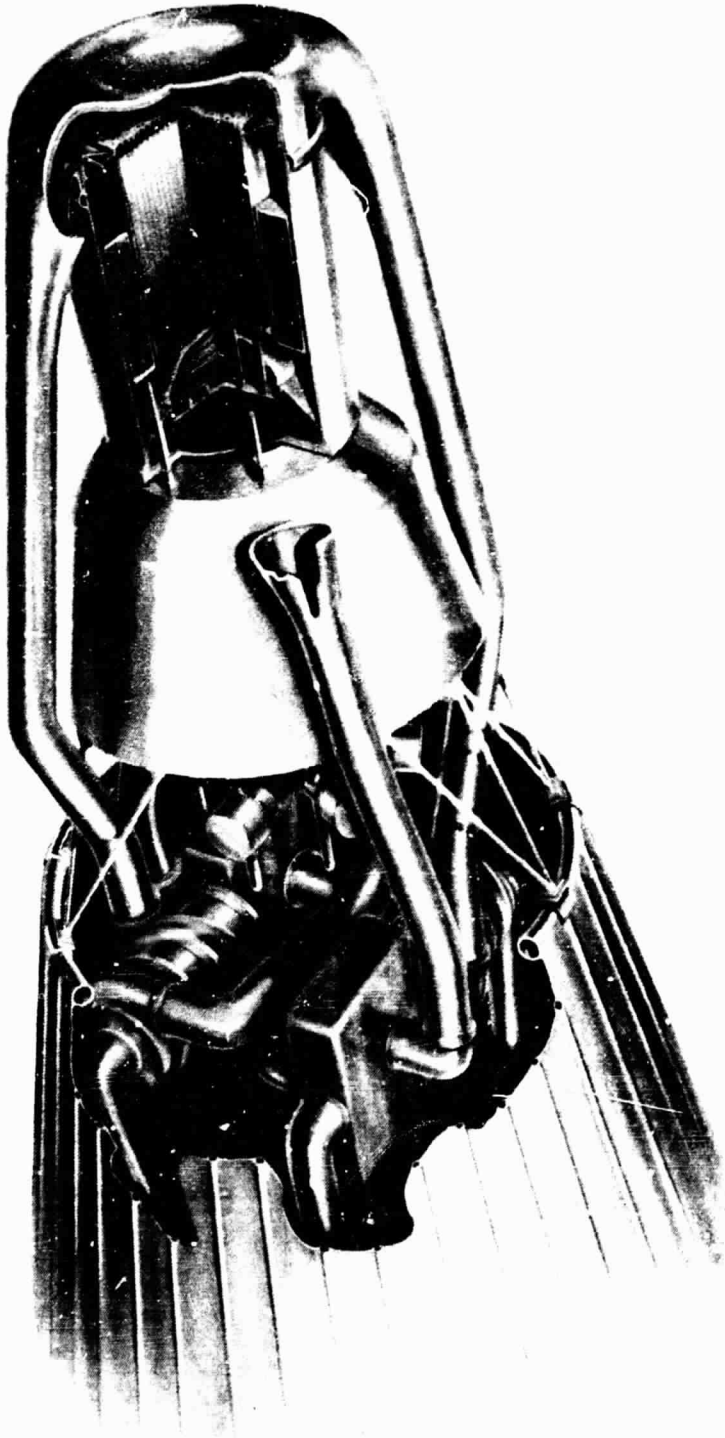
FIGURE 1   Gas-cooled, metal-clad reactor system.

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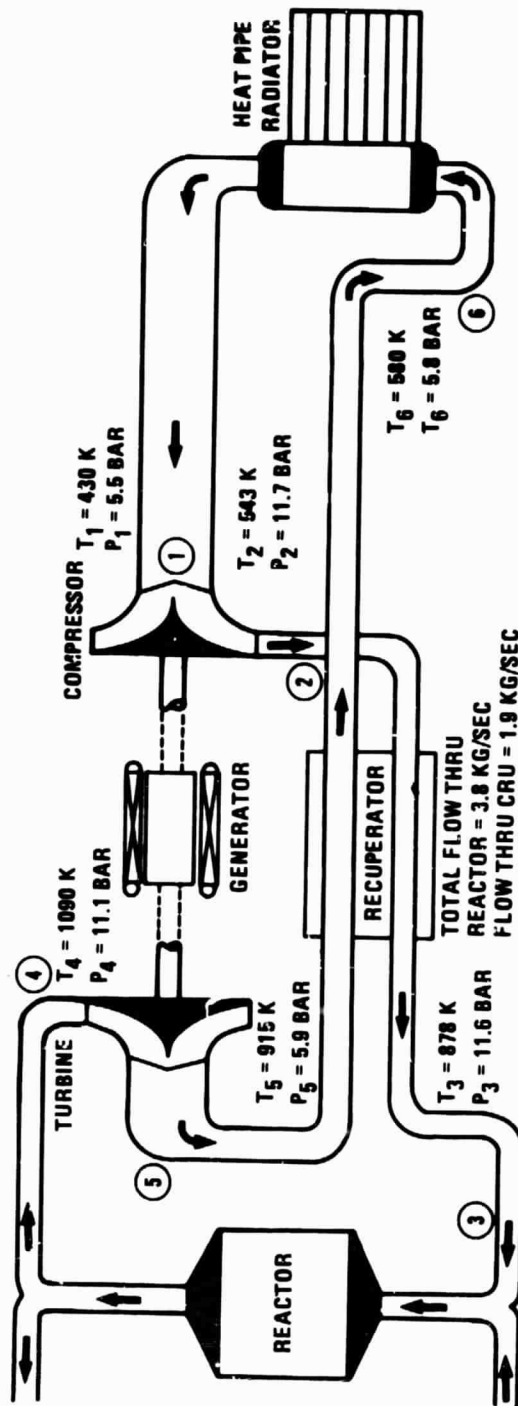
FIGURE 2   The 100-kW(e) reference cycle.
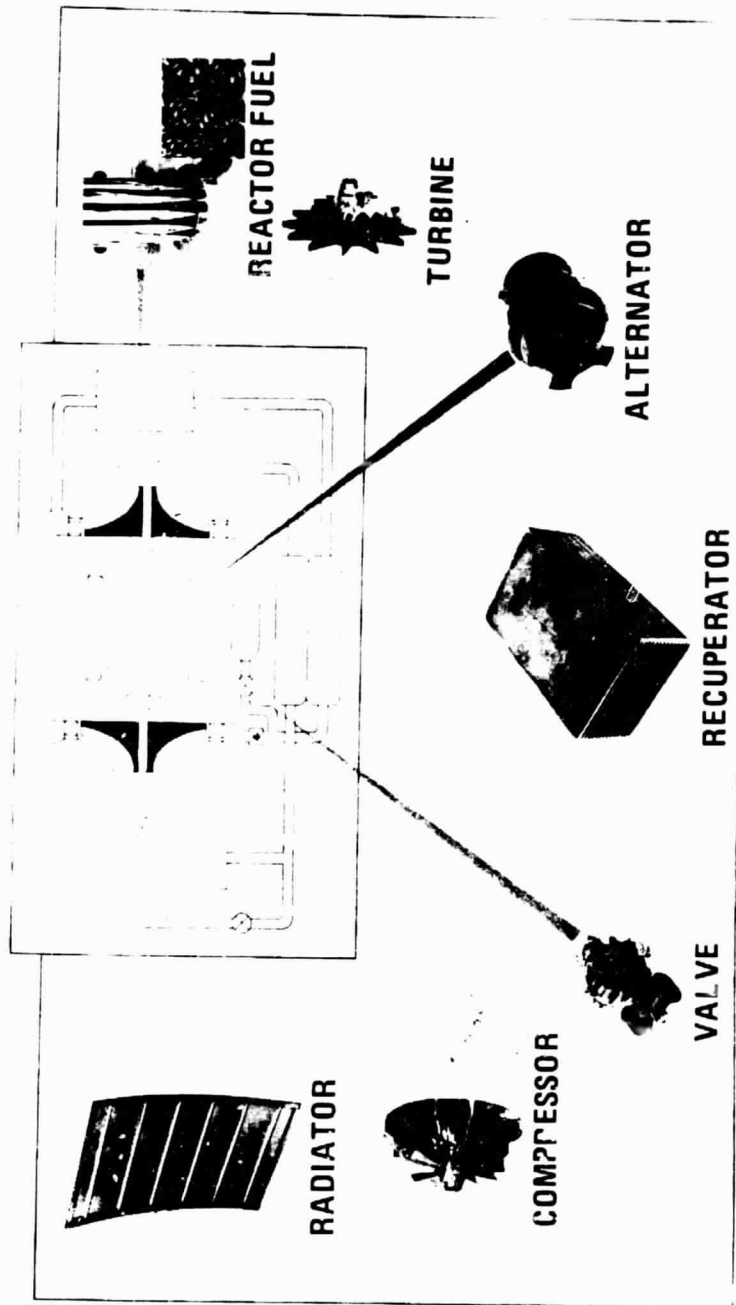
FIGURE 3   Typical Brayton-cycle components (Garrett Corporation, 1979).

FUEL ELEMENT

FUEL ROD

FISSILE
(U-235
OR U-233)

FERTILE
(Th-232)
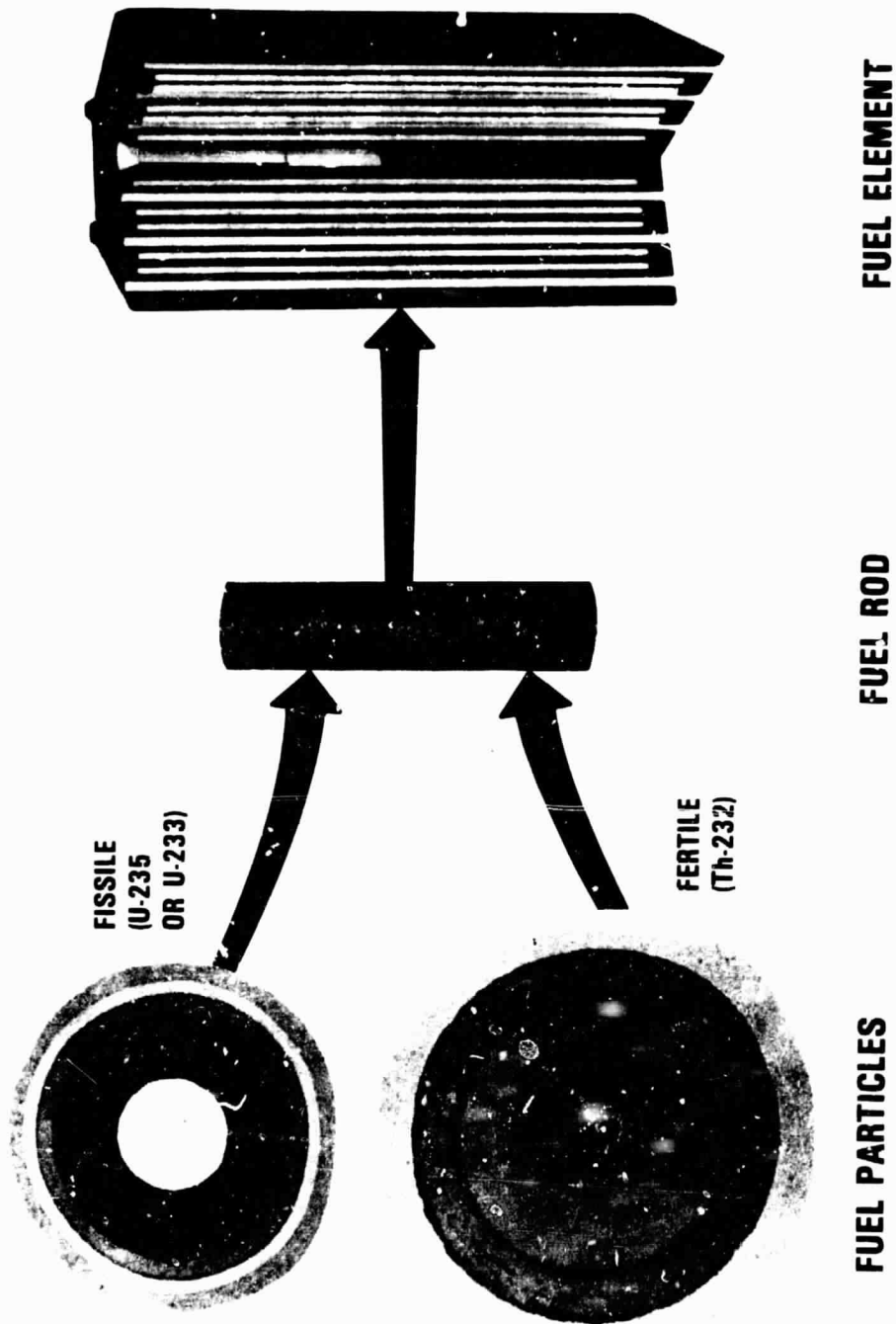
FUEL PARTICLES

FIGURE 4 HTGR fuel components.

ZrC COATED COOLANT CHANNEL

200 CM

1.9 CM
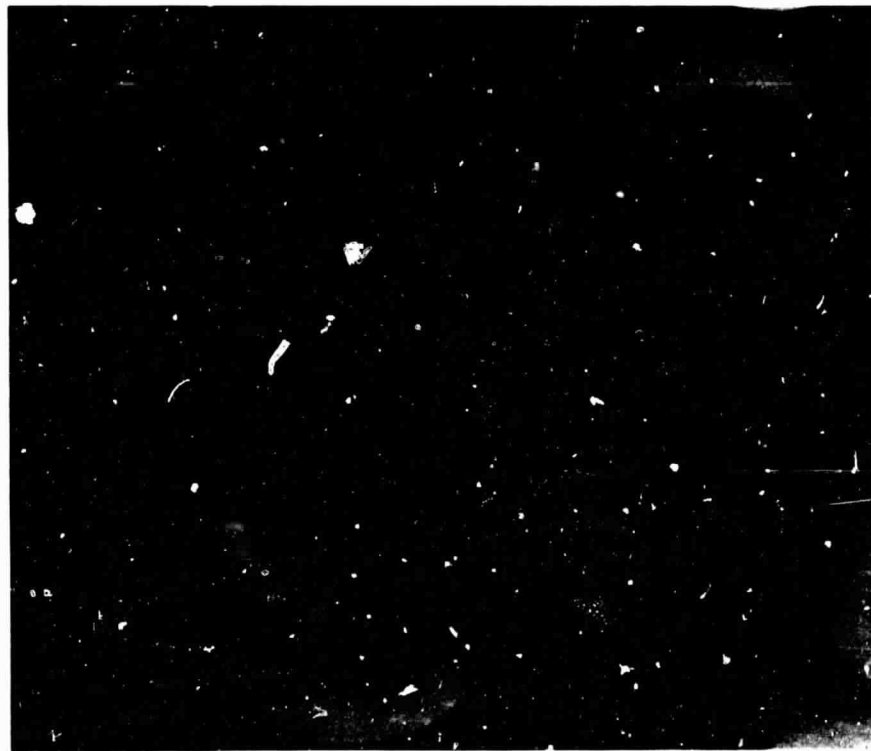
FUEL ELEMENT

COATED FUEL PARTICLES
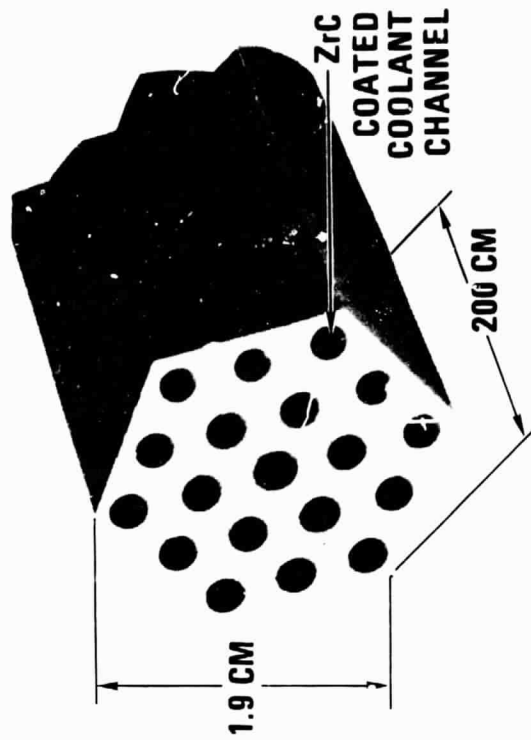
FIGURE 5   NERVA fuel.
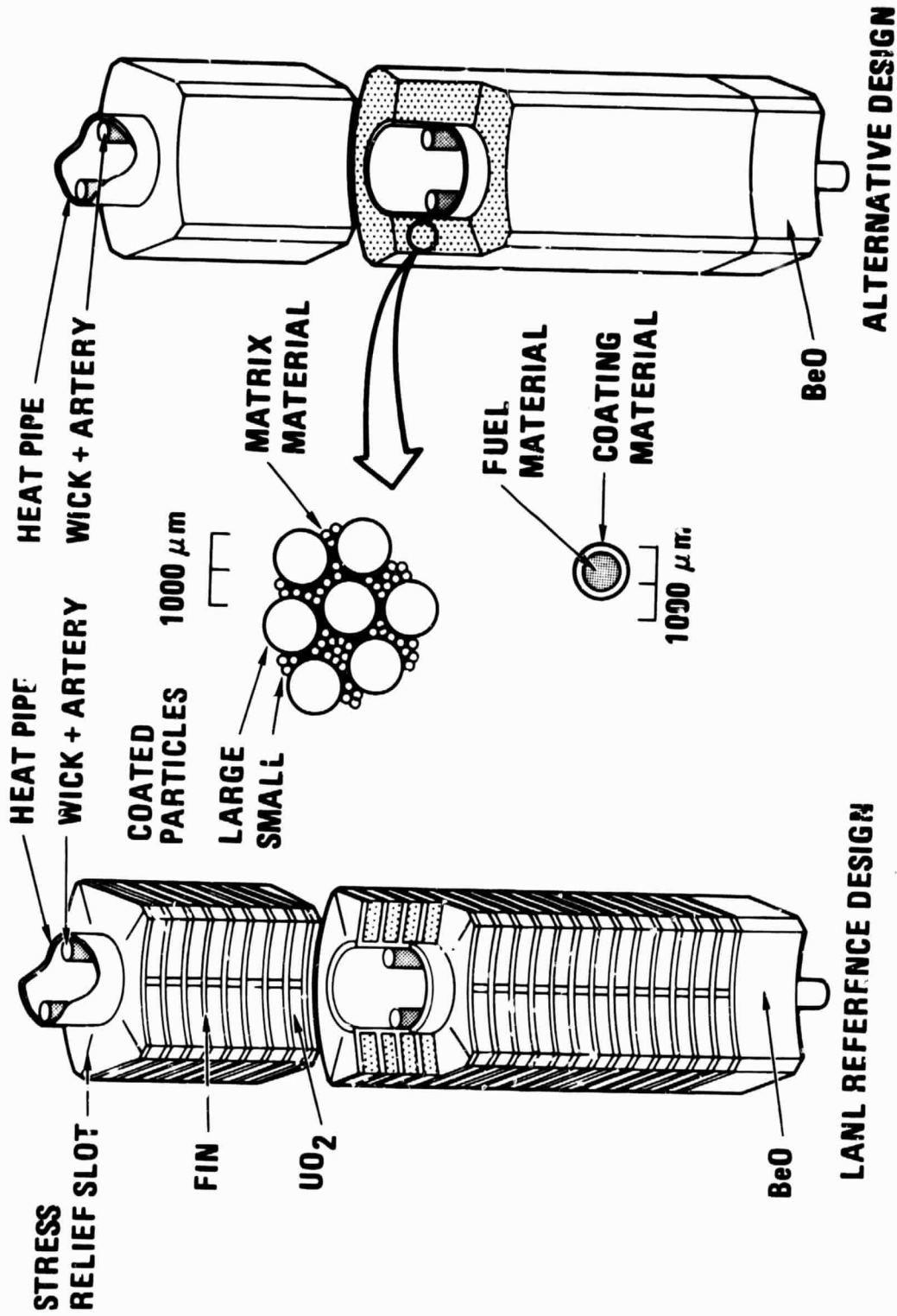
FIGURE 6 Alternative SP-100 fuel design.
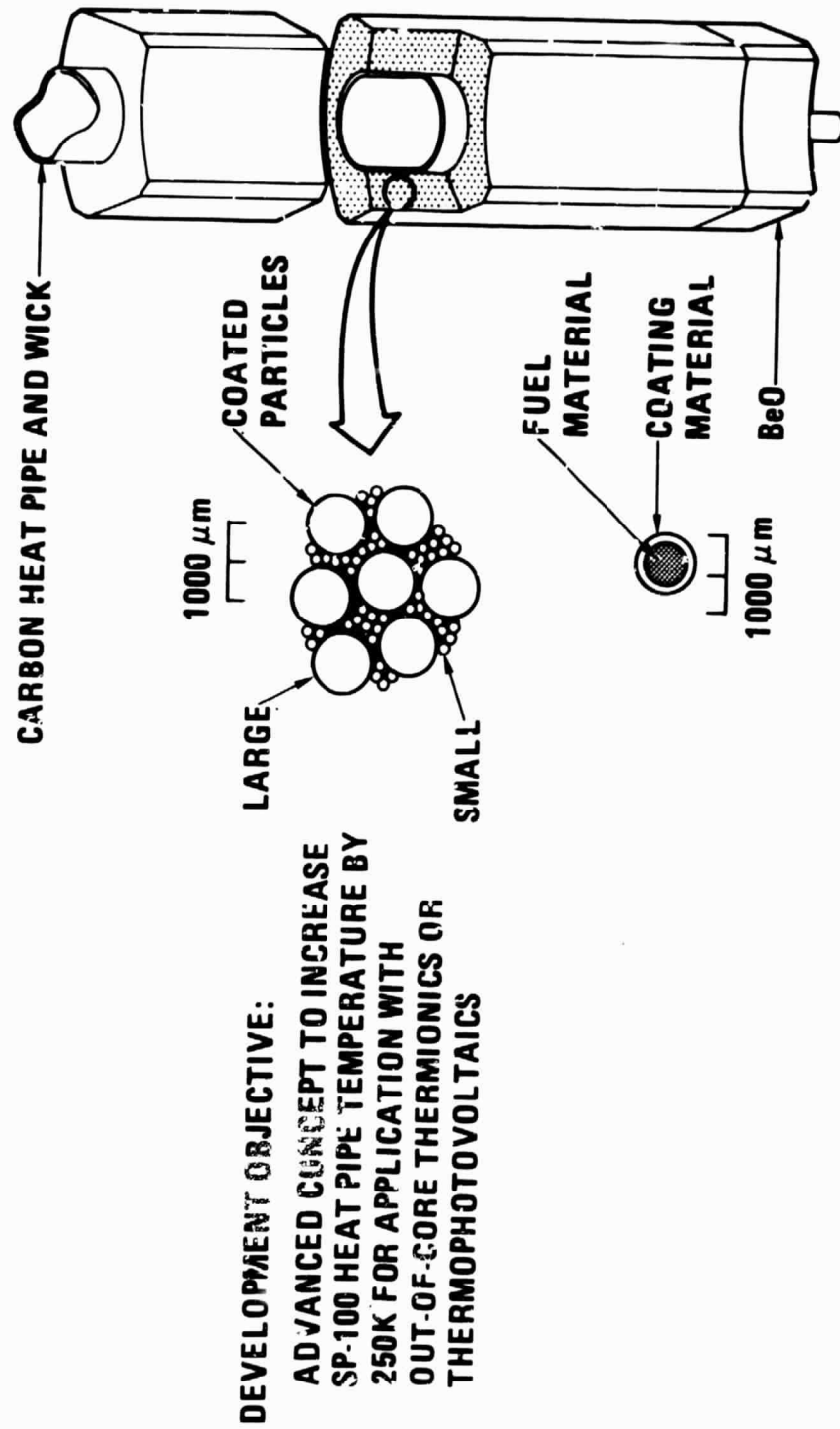
FIGURE 7 Advanced SP-100 reactor using all-carbon system.

HIGH STRENGTH
GRAPHITE CLAD

DEVELOPMENT OBJECTIVE:

ADVANCED CONCEPT TO OBTAIN GAS
COOLED REACTOR OUTLET TEMPERATURE
OF 1500K FOR USE WITH A CERAMIC
TURBINE-BRAYTON CYCLE

COATED
PARTICLES

LARGE

SMALL

1000 μm

FUEL
MATERIAL

COATING
MATERIAL

1000 μm
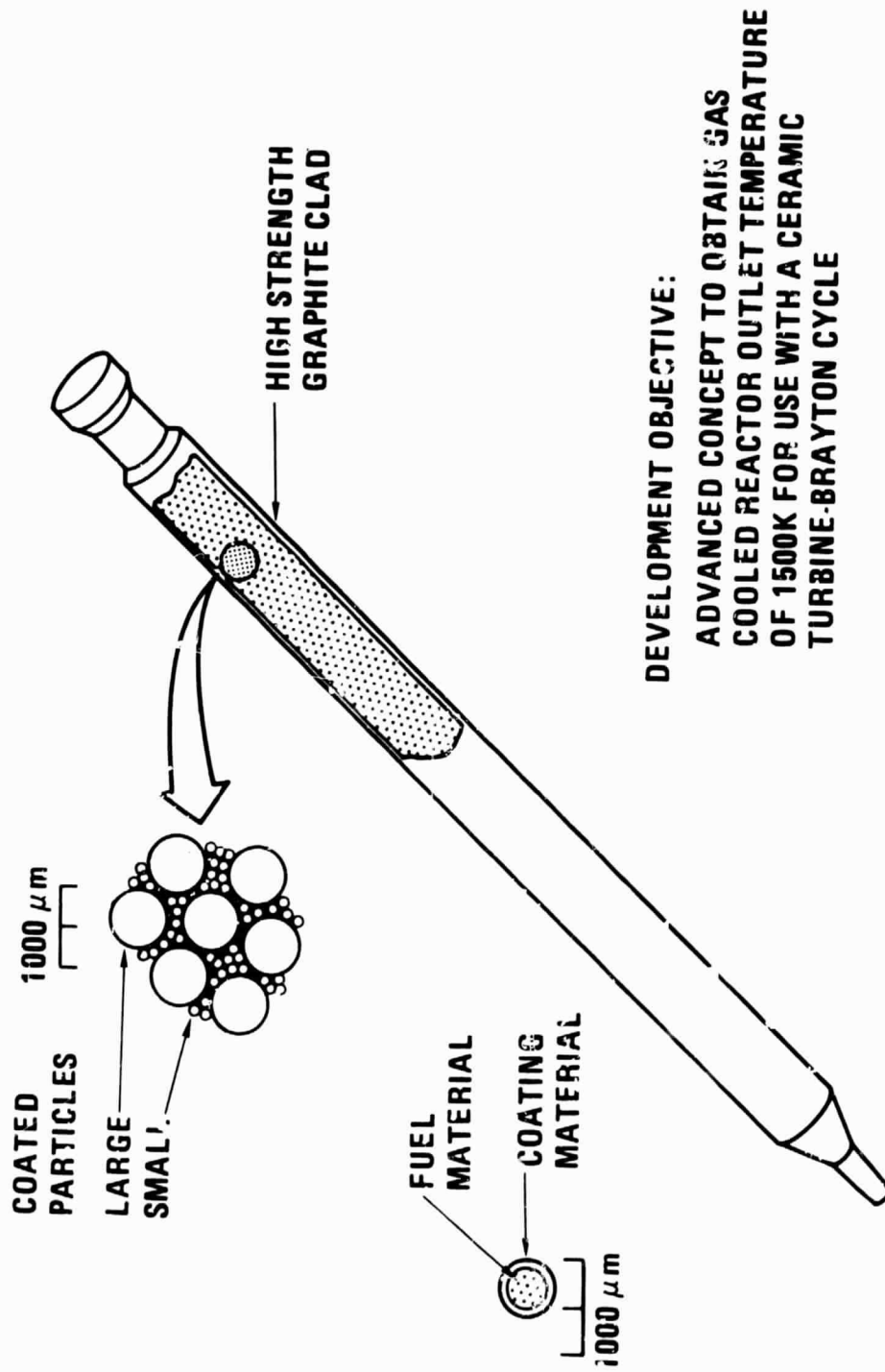
FIGURE 8 Gas-cooled, graphite-clad fuel.

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CURRENT-VOLTAGE CHARACTERISTIC

POTENTIAL DIAGRAM

THERMIONIC CONVERTER SCHEMATIC
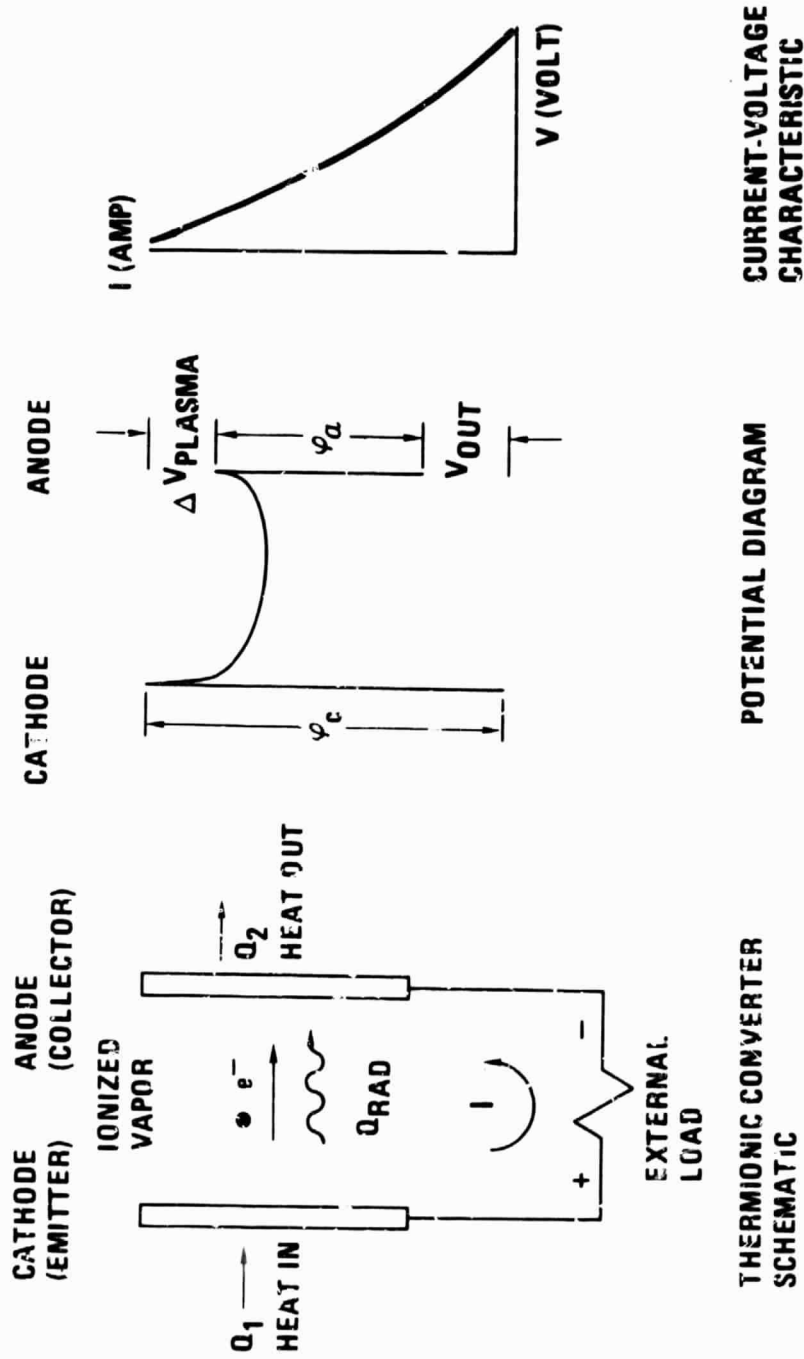
FIGURE 9   Thermionic energy conversion.

FIGURE 10   Thermionic cell.

FIGURE 11   Thermionic fuel element (TFE).

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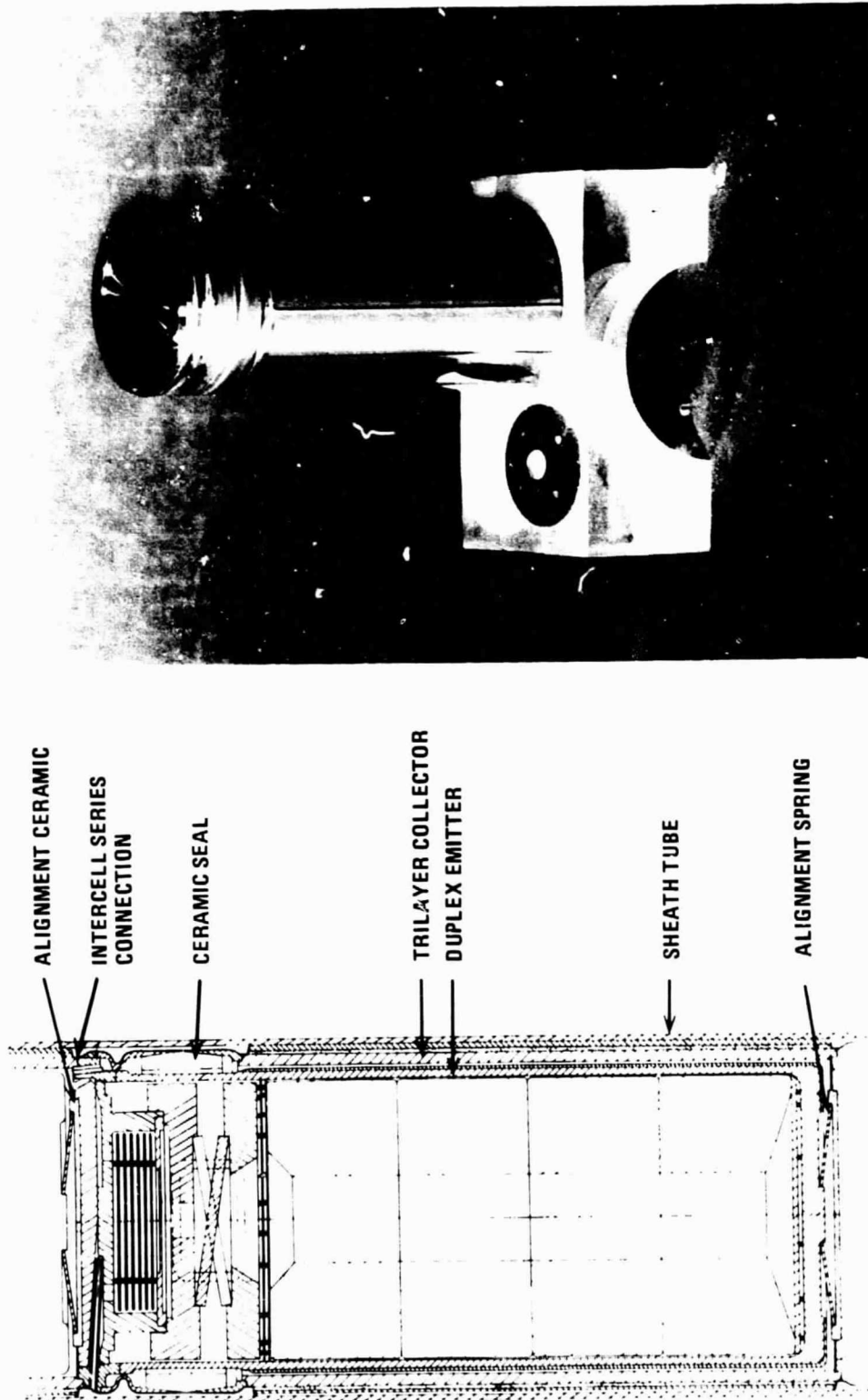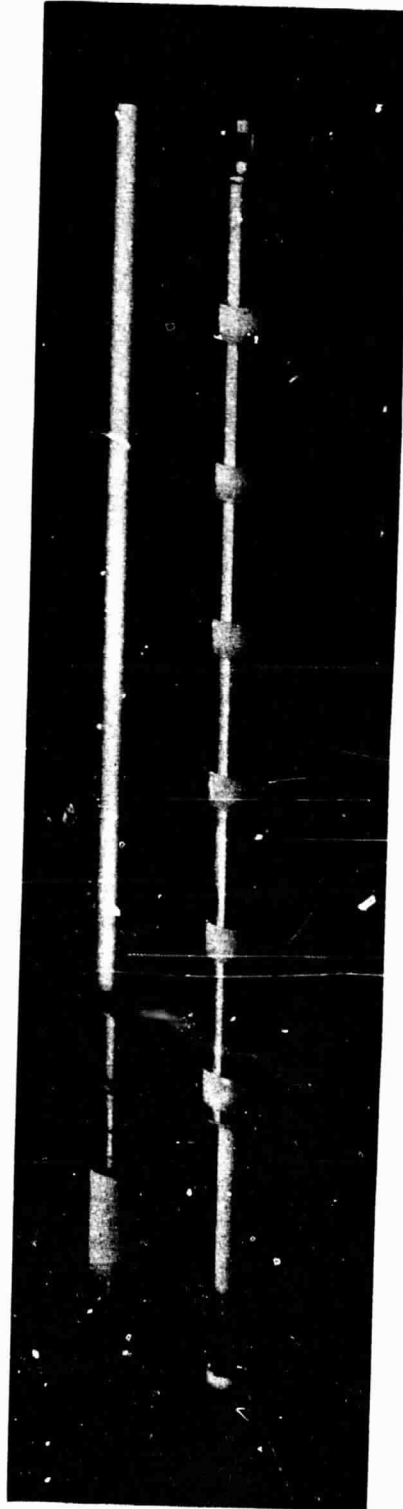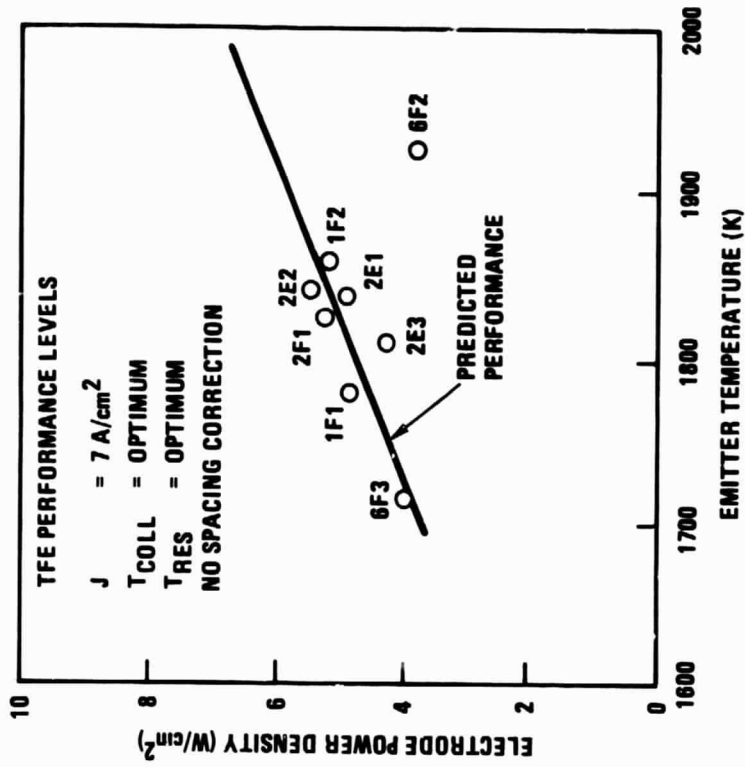
TEST SERIES	NUMBER OF CELLS IN TFE	NUMBER OF TESTS	MAXIMUM TEST HOURS
IN-PILE			
F	1	4 (1)*	8,560
F	2	3 (1)	8,020
F	6	5 (3)	8,062
E	2	3 (1)	12,535
OUT-OF-PILE			
ELECTRICALLY HEATED	–	4 (2)	46,647

*( ) ASSEMBLIES UNDER TEST AT PROGRAM TERMINATION.

FIGURE 12 Thermionic test assemblies and performance comparison.

90
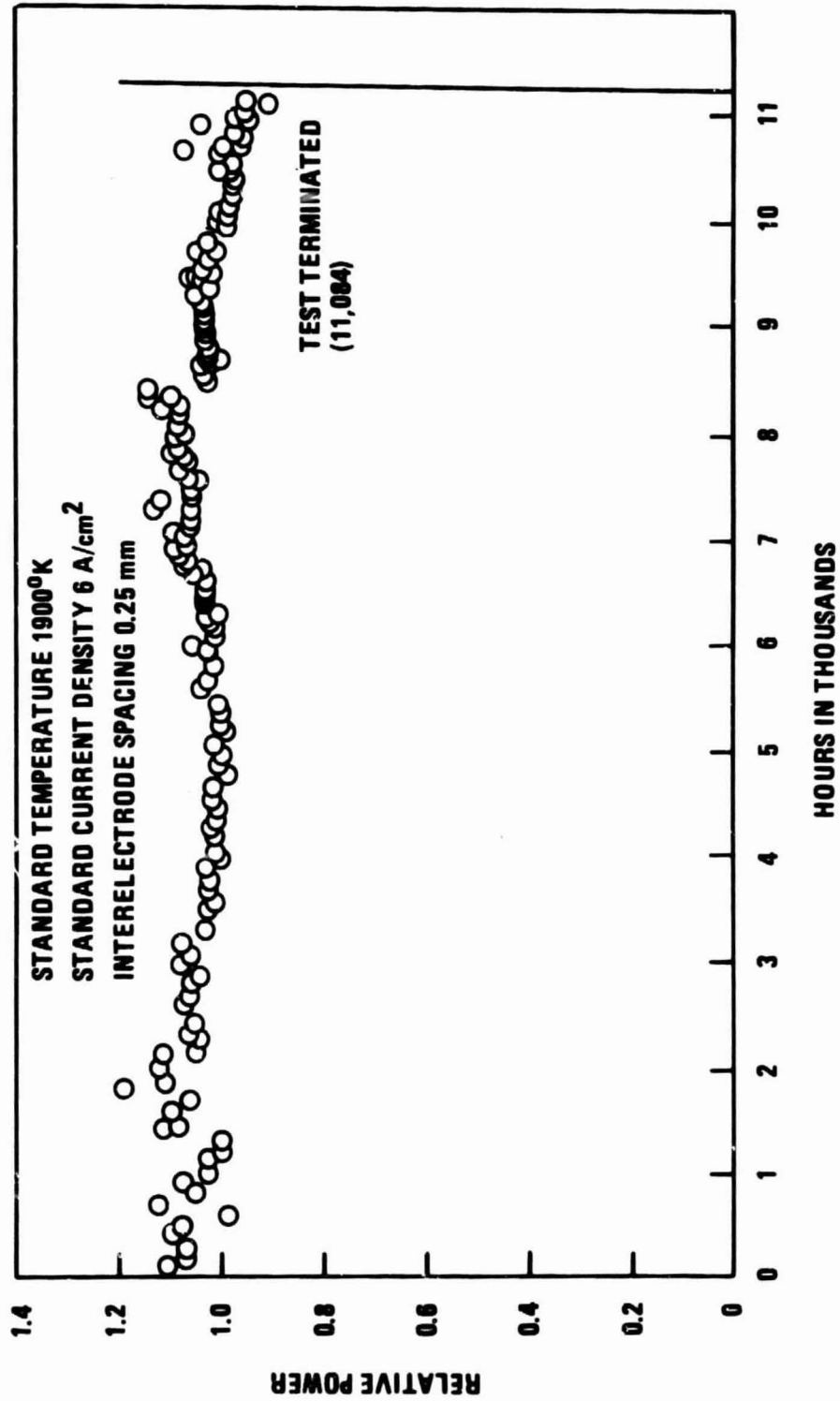
FIGURE 13 TFE 2E2 relative performance history.
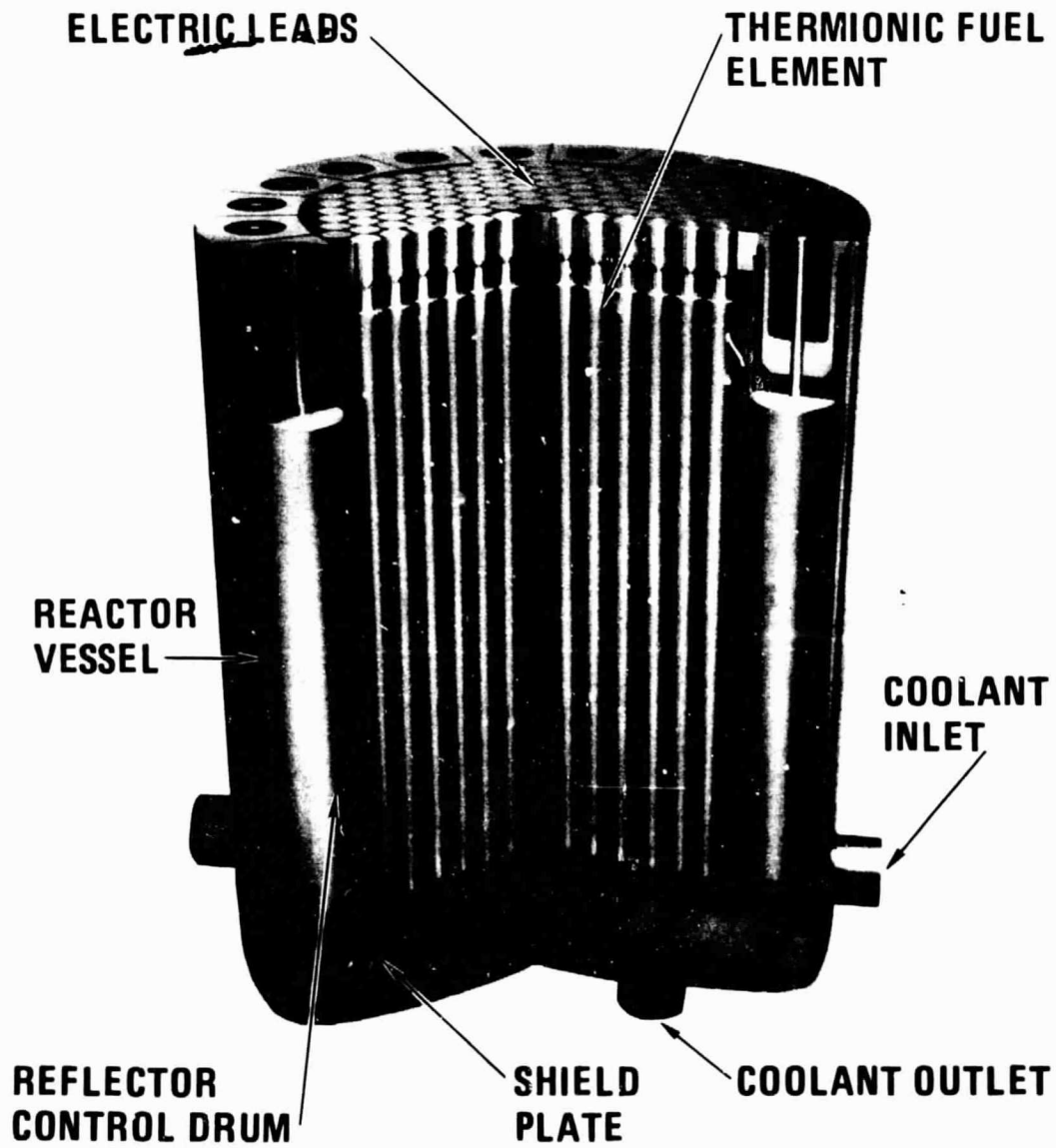
FIGURE 14 Thermionic reactor.

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THERMIONIC FUEL ELEMENT (TFE) DEVELOPMENT

TFE

APPLICATIONS

10 – 12 kWe
FLIP TRIGA

5 – 10 kWe
MODERATED
UZrH DRIVER

40 – 60 kWe
MODERATED
UZrH DRIVER

100 kWe – 1 MWe
FAST (U235)
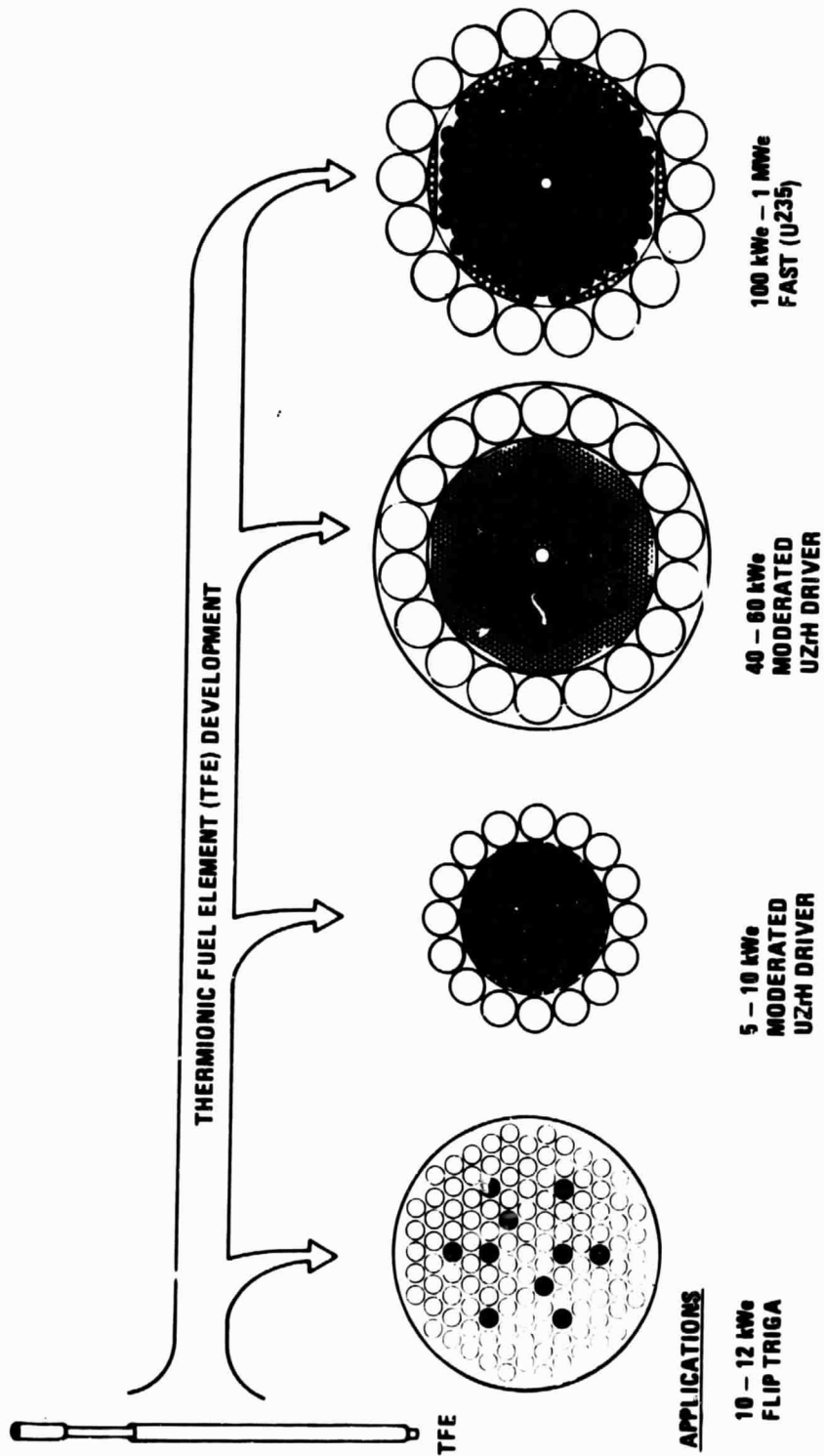
FIGURE 15 Thermionic reactor configurations.
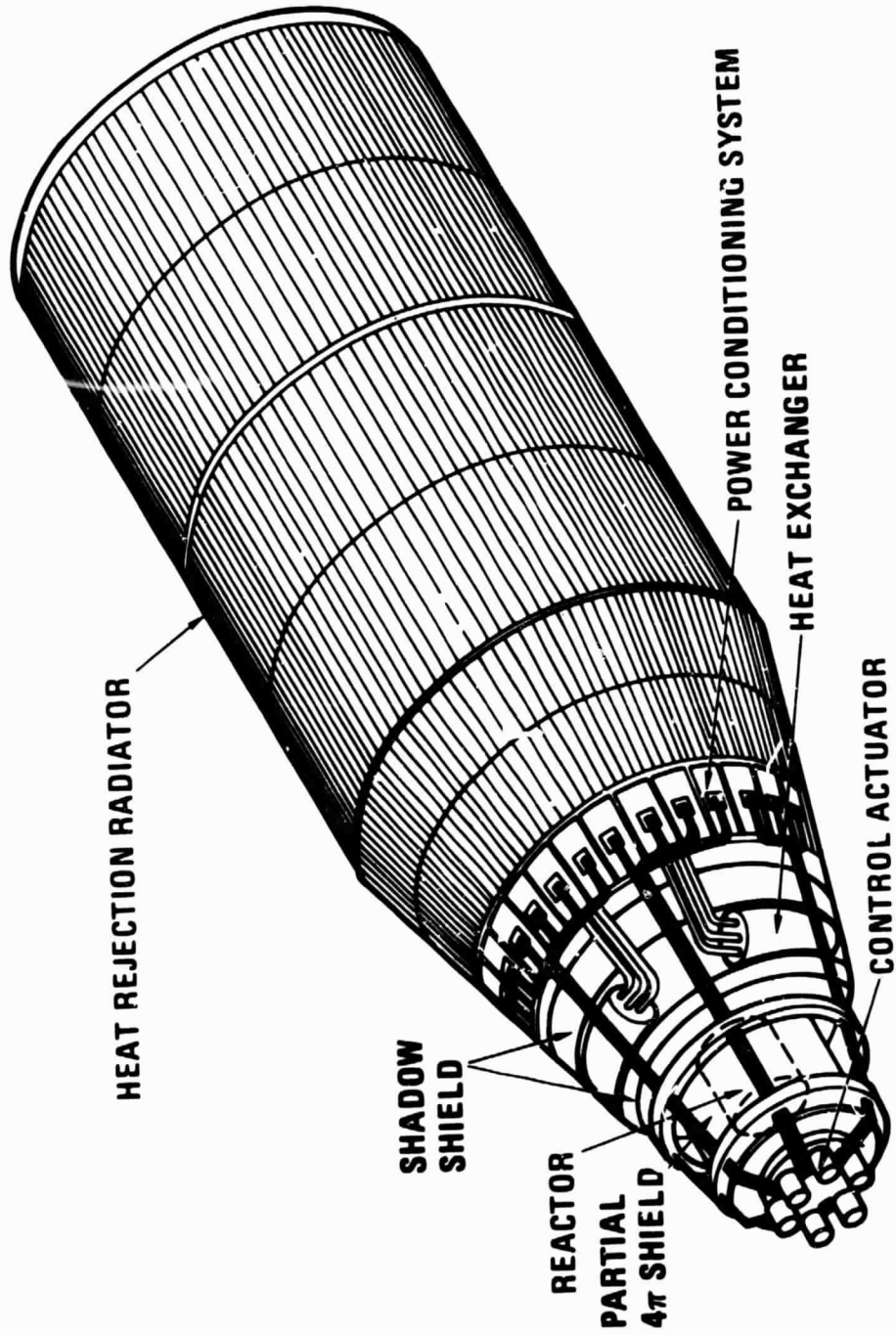
FIGURE 16 Thermionic reactor power system.

94
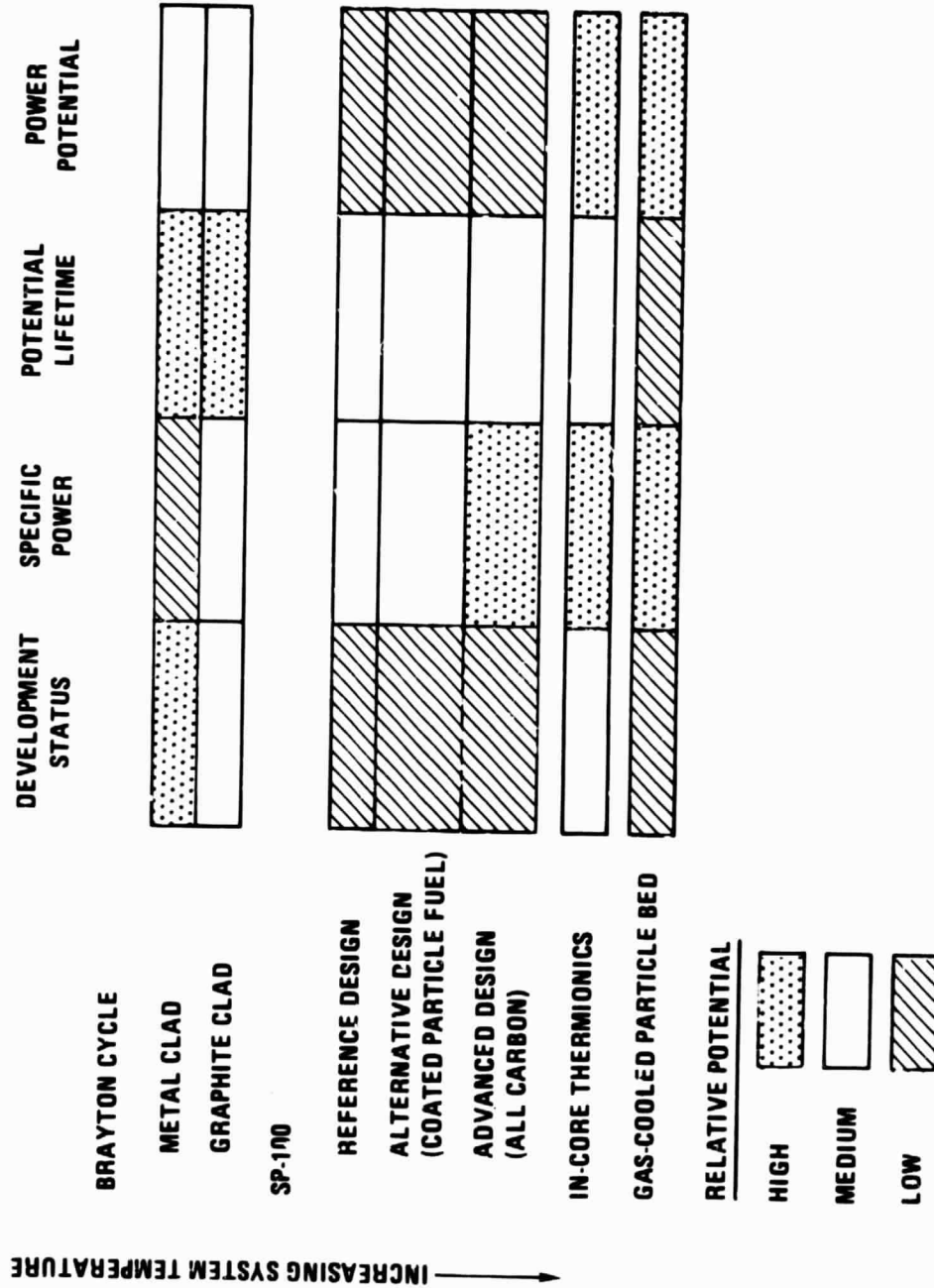
FIGURE 1.7 Qualitative comparison of concepts relative to SP-100 reference design.

PARTICLE BED REACTORS AND RELATED CONCEPTS

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ABSTRACT

Compact high-performance reactor designs based on HTGR particulate
fuel are analyzed. The large surface area available with the
small-diameter-particulate (about 500 μm) fuel allows very high power
densities (megawatts per liter), small temperature differences between
fuel and coolant (about 10 K), high coolant outlet temperature
(1500-3000 K, depending on design), and fast reactor start up (about
2-3 s). Two reactor concepts are developed--the Fixed-Bed Reactor
(FBR), where the fuel particles are packed into a thin annular bed
between two porous cylindrical frits, and the Rotating-Bed Reactor
(RBR), where the fuel particles are held inside a cold rotating
(typically about 500 rpm) porous cylindrical frit. In both designs,
coolant flows through the fuel bed in the radially inward direction.
The FBR can operate for long periods (months to years) in the
closed-cycle He-cooled mode or in the open-cycle H_2-cooled mode. In
the latter mode, integrated operational time is a few hours at most,
owing to H_2 storage limitations. The FBR can quickly switch between
modes, if desired. The RBR will operate only in the open-cycle
H_2-cooled mode. Maximum power capability for the FBR is about 300
MW(t) in the closed-cycle mode and about 1000 MW(t) in the open-cycle
mode; maximum power for the RBR is about 5000 MW(t). The FBR would be
used with a turbine in either the closed-cycle or the open-cycle mode;
the RBR could be used for direct thrust or could produce electricity
with a magnetohydrodynamic (MHD) generator or a turbogenerator.
Experiments on particle beds relating to the FBR and RBR are
described. Development requirements are assessed. These appear
modest for the FBR but are more substantial for the RBR.

INTRODUCTION

The increasing activity in space will require major advances in power
and propulsion. This paper describes reactor systems being

investigated at Brookhaven National Laboratory (BNL) based on the small-diameter-particulate fuel developed for high-temperature gas-cooled reactors (HTGRs). These reactors have high-performance potential and appear suitable for relatively quick development.

Figure 1 illustrates the spectrum of potential power systems for space nuclear power. At the low end of the power spectrum (hundreds of kilowatts to a few megawatts), electrical generation will typically be for long periods (months to years). This dictates a closed thermodynamic cycle, with radiative heat rejection to space. These reactors would be used for station keeping, communications, radar, etc.

At the other end of the power spectrum, reactors with power outputs of tens to hundreds of megawatts are desired. These would generate electric power for pulsed-energy devices or act as direct thrust nuclear rockets for orbit raising. Such reactors probably will operate in the open-cycle mode with high-temperature H_2 coolant. They will have a relatively short integrated operating life (minutes to hours). Heat rejection would be provided by dumping waste coolant into space. For very short lifetime generation, open-cycle chemical systems (H_2/O_2 with turbines or MHD generators) could be considered. However, H_2-cooled reactors can generate much more electrical energy per kilogram of coolant consumed than chemical systems. Nuclear systems are thus preferred for operating times longer than a few minutes.

Reactors could also operate at intermediate powers, generating electric power for electric thrusters or advanced pulsed-energy systems. The long operating time in these applications will require closed-cycle power generation. At the power levels shown, a lightweight radiator will be necessary.

An attractive lightweight liquid-droplet radiator (LDR) has been proposed by Mattick and Hertzberg (1980). In this concept, reject heat would radiate from a sheet of small-diameter liquid drops (e.g., 100-μm lithium drops). The sheet would be sprayed from a set of nozzles toward a collector. The collected liquid would be reheated and recycled to the spray generator. The LDR should be a very light structure compared with conventional fin-tube or heat pipe radiators. If developed, heat rejection at powers of tens of megawatts would prove practical.

Assuming a high-performance LDR, closed-cycle systems could deliver electric powers comparable to those of open-cycle systems. Powers up to about 10 MW(e) could be used in electric thrusters. For advanced energy applications, cw (continuous wave, i.e., steady state) powers of tens of megawatts (electric) are desired.

Multimodal operation is more desirable than single-mode operation. Having one reactor carry out different missions eliminates the weight of extra reactors. Figure 2 shows two examples of multimode operation. In the bimodal system, the reactor operates most of the time in a closed-cycle (cw) mode at relatively low power (Mode 1). When necessary, it rapidly shifts to a high-power, open-cycle mode (Mode 2). Such a reactor could continuously generate hundreds of

kilowatts for station keeping, communications, radar, etc. Upon
demand, it could supply tens to hundreds of megawatts for short
periods (minutes to hour) for pulsed-power applications.

In the trimodal system, the reactor would operate in three modes:
(1) a short-term, high-power, direct thrust mode for orbit raising,
(2) a long-term, closed-cycle, low-power mode for station keeping,
etc., and (3) a short-term, high-power, pulsed electric generation
mode.

The advantages of multimode operation are somewhat offset by
increased system complexity. For any particular mission, the
trade-offs between lower weight and increased complexity will
determine whether separate or multimode operation is more desirable.

The time and costs to develop any space reactor will be
substantial. Thus there is a strong incentive to develop space
reactors that will cover as wide a range of applications as possible.

The compact reactor concepts based on HTGR particulate fuel being
developed at BNL should meet a wide range of applications. They range
from the low-power (hundreds of kilowatts) to the high-power (hundreds
of megawatts) end of the spectrum, in either open-cycle or
closed-cycle mode. The high-power-density and high-temperature
capabilities of particulate fuel result in high-performance systems.

PARTICULATE-FUELED REACTOR CONCEPTS

The BNL compact reactor concepts are based on small-diameter-
particulate (about $500\ \mu m$) fuel that has been developed and used in
high-temperature, gas-cooled reactors in the United States and abroad
(e.g., the U.S. HTGR, the German HTGR). The reasons for this choice
are the following:

1. The fuel is available
   o commercially fabricated in HTGRs
   o well characterized.

2. Coolant outlet temperature is very high
   o 3000 K for open-cycle H_2
   o 1500 K for closed-cycle He.

3. Reactor power performance is excellent
   o up to approximately 10-MW/ℓ power density in fuel
   o approximately 10-K temperature difference between fuel and
     coolant
   o potential reactor outputs up to thousands of megawatts.

4. Materials behavior is excellent
   o no thermal shock to fuel
   o full power in seconds

o retention of 99.99 percent of fission products
o burnups of more than 50 percent possible.

Figure 3 shows the BISO and TRISO fuel particles developed by General Atomics for the HTGR (Gulden and Nickel, 1977). The BISO particles have a coating of pyrographite over a kernel of fissile or fertile material and are suitable when the internal pressure of gaseous fission products is small. (The inner porous low-density layer holds gaseous fission products.) BISO ThO_2 particles are used in the HTGR for in situ breeding of fissile ^{233}U.

TRISO particles have a multilayered structure and achieve high burnups at high internal pressures of gaseous fission products. They have a fissile kernel of $^{235}UO_2$ or $^{235}UC_2$ (or mixed oxycarbides), with a porous low-density pyrocarbon layer to hold gaseous fission products. The SiC layer provides mechanical containment of the internal pressure. Burnups of more than 50 percent of the fissile inventory are possible, as well as excellent fission product retention (99.99 percent).

Commercially available HTGR fuel is suitable for the closed-cycle applications considered in this study. Open-cycle operation will require some modification of the fuel to prevent attack by H_2 coolant. (The fuel particles would be coated with ZrC, for example.)

The small diameter of the fuel particulate results in very large internal heat transfer area per unit volume of fuel. A packed bed of 500-μm particles, for example, has an internal surface area of 100 cm^2/cm^3 of bed. This large heat transfer area allows very high power densities and small ΔT's between coolant and fuel. The small diameter of the fuel particles minimizes thermal stress and thermal shock. In conventional reactors, the rate of change of temperature of the much larger fuel elements must be slow in order to prevent fuel damage. Time to full power in these reactors is thus long (minutes to hours, depending on design). In contrast, the BNL compact reactors should have rise times of a few seconds.

Two types of particulate fuel reactors are being investigated at BNL. In the first, the Fixed-Bed Reactor (FBR), the fuel is held between two porous "frits," as shown in Figure 4. The "frit" can be made of metal or ceramic and has substantial strength. The pressure drop across the frits is comparable to the drop across the fuel bed. Frits are fabricated by sintering wires or particles into cylindrical or platelike structures.

Gaseous coolant (i.e., He or H_2) enters the inlet frit, passes through the bed, and exists from the outlet frit. The packed bed is typically several centimeters in thickness. The fuel particles in the FBR are held in place and do not move.

In the second type of particulate fuel reactor, the Rotating-Bed Reactor (RBR), the fuel particles are held by centrifugal force inside a rotating cylindrical metal frit (Figure 4). Cold gas enters the frit and is heated as it passes through the fuel particle bed. The

hot gas then enters the internal cylindrical cavity and exhausts through a nozzle (not shown) at the end of the cavity.

The rotating bed can operate in any of three modes: (1) fully settled (all particles held against the rotating frit), (2) fully fluidized (all particles suspended by gas flow), and (3) partly fluidized (Figure 4). In the partly fluidized mode, gas velocity in the cool outer part of the bed is insufficient for fluidization. At some point inside the bed, the temperature and velocity of the coolant become high enough to fluidize the inner portion of the bed.

In all three modes, the fuel particles are fully contained. The fluidized portion acts like a dense fluid, with gas passing between the suspended particles. Particles cannot escape from the bed unless it is allowed to expand to the point where adjacent particles do not interact. Bed operating parameters can be chosen to prevent such "flooding."

For any given set of conditions (flow rate, reactor size, etc.), frit rotation speed controls the fluidization regime. As rotation decreases, the bed first changes from the fully settled to the partly fluidized mode and then to the fully fluidized mode. These operating modes have been demonstrated in experiments described later.

Because the RBR has no high-temperature frit and a lower pressure drop than the FBR (for the partly and the fully fluidized modes), its outlet temperature and power can be considerably greater than those of the FBR.

FBR and RBR power densities can be very high. Figure 5 shows the pressure drop in FBR and RBR beds as a function of average bed power density (in MW/l) for typical operating conditions (pressure, 100 atm; diameter, 500 μm; bed thickness, 4 cm).

The curves show three different coolant/temperature regimes. Closed-cycle, long-duration FBRs would use He coolant at relatively low outlet temperatures (1300-1500 K) because of long-term turbine temperature limitations. Open-cycle FBRs would use H_2 to minimize the coolant mass dumped into space. The short operating time (minutes to hours) will allow higher outlet temperatures with shorter life turbines. Outlet temperature will depend on materials used in the inner frit as well as turbine capabilities. Depending on design, outlet temperatures in the range from 1500 to 2500 K appear practical. Open-cycle RBRs can operate to higher outlet temperatures (up to about 3000 K) because they are not restricted by a high-temperature frit. They could provide direct thrust propulsion or could generate electric power with MHD generators or high-temperature turbines.

As is shown in Figure 5, power densities of 10 MW/l and higher could be achieved with the FBR and RBR. The large heat transfer area in the bed makes local temperature differences between the fuel and gas coolant very small. Figure 6 shows average film ΔT between fuel particles and the gas coolant as a function of power density, for the conditions shown in Figure 5.

Film ΔT is only a few degrees Kelvin, even at very high power densities. The ΔT inside the fuel particles (center to the surface) is even less, about 1 K. Thermal stress in the particles is very low, less than 1 MPa. The thermal diffusion time is small, approximately 5 ms for 500-μm particles. This is much shorter than the ramp time for power or coolant flow changes. Thermal shock or fatigue damage to the particles appears unlikely.

Particle beds can ramp from zero to full power in a few seconds. Experiments have demonstrated that electrically heated beds can reach full power in a couple of seconds. The beds can be recycled repeatedly with no damage to the particles. In contrast, reactors with rod or plate fuel elements cannot rapidly ramp to full power because of thermal shock problems. (NERVA took over 1 min, for example.)

Two possible FBR fuel bed configurations are shown in Figure 7. Figure 7a shows the particles packed between two annular cylindrical frits. Coolant flows radially through the annular bed from an adjacent inlet plenum to an outlet plenum. In Figure 7b, the particles are packed into small annular fuel elements between two frits. These elements are spaced in a moderator matrix and held by a grid plate. In each element, coolant flows radially through the packed bed that is between adjacent plenums.

The first type of FBR is simpler and is preferred for most applications. The second type can operate at higher power densities and use a hydrogenous moderator, such as ZrH_2. (The first type will not go critical with a hydrogenous moderator.) It could be used when very small size is desired.

The first type of FBR (Figure 8) can generate up to about 1,000 MW(t) with H_2 coolant and about 300 MW(t) with He coolant. Reactor size scales with power level. Minimum fuel bed diameter is about 30 cm, from considerations of criticality and burnup. For maximum power, fuel bed diameter is approximately 80 cm. (Larger diameters are possible, but a more compact configuration would probably be used if higher powers were desired.) Thickness of the annular fuel bed varies with diameter and ranges from about 3 to about 8 cm.

The moderator/reflector zones in the FBR (Figure 8) slow down fission neutrons, reflecting them back into the fuel, where they are absorbed. FBR neutronics are discussed in more detail later. Beryllium moderator/reflectors result in the lowest critical masses. The reflector should be at least 25-30 cm thick. Thicker reflectors are marginally better, while thinner reflectors leak too many neutrons. Graphite is a good internal moderator/reflector and is compatible with the high-temperature coolant.

The FBR is controlled by movable drums or rods in the external moderator/reflector. The control drums (Figure 8) have a neutron absorber (e.g., B_4C) on one side and a moderator on the other. Rotation of the drum moves the poison relative to the fuel. Movement toward the fuel decreases k_{eff}; movement away increases it.

The high importance of neutrons in the external moderator/reflector makes poison control very effective. Also, externally moderated reactors have very long neutron lifetimes. This makes the FBR neutronically "sluggish," so that even very large reactivity insertions would cause only a relatively slow rate of power rise. Both factors make the FBR easier to control than conventional power reactors.

Depending on design, the critical mass of the FBR will be between 10-40 kg of fully enriched uranium (93.5 percent ^{235}U). With additional fuel for burnup, the FBR could operate in the range of about 50-100 MW(t). Critical masses for RBRs are about twice those of FBRs because RBRs do not have an internal moderator. Peak/average power ratios are also higher for RBRs.

The RBR (Figure 9) would operate only in the open-cycle H_2-cooled mode, with a maximum integrated operating life of a few hours. H_2 mass requirements preclude longer operation. Exit coolant temperature would be limited by fuel lifetime, not structure. Tests of ZrC-UC fuel particles in H_2 indicate that they could operate for several hours at 3000 K.

Equilibrium MHD generators need very high temperatures. Practical generators operating on H_2 require 3000 K for good performance. Direct thrust for orbit raising also needs very high temperatures. (The specific impulse of H_2 is about 1,000 s at 3000 K.) Lower temperatures significantly degrade specific impulse.

Turbines give good performance without having to operate at 3000 K. Although increasing temperature improves performance, the gain probably does not warrant developing extra high temperature turbines. Hydrogen-cooled turbines operated successfully in the early 1960s at turbine inlet temperatures of about 2500 K. Operating RBRs with turbines at these temperatures yield very attractive efficiencies. Fuel particle lifetime would also be much longer at 2500 K.

Operating regimes for FBRs and RBRs are shown in Figure 10. The boundaries can shift, depending on design. For example, FBRs could deliver higher outlet temperatures with a ZrC ceramic frit. Also, maximum power for the FBR can be increased by design changes.

THE FBR (FIXED-BED REACTOR)

Scoping studies of FBR neutronics have been carried out with the one-dimensional ANISN transport code. A 15-group structure was used (7 thermal groups), with collapsed cross sections derived from the ENDF-B-V file, and a P_1S_8 scattering kernel.

Parameter k_{eff} was calculated as a function of ^{235}U loading for infinite cylinders. Critical masses for finite-cylinder FBRs (L/D = 1) were approximated by assuming that they had the same fissile loading as k_{eff} = 1.1 infinite cylinders (for sections equal in length to the finite cylinder). End effects for FBRs are expected to be small. The fuel bed diameter is much greater than the bed thickness, and the

inner and outer moderator/reflectors fit closely. End reflectors will affect only the last few centimeters of the fuel bed. The 10 percent allowance in ΔK for end effects thus seems quite conservative.

Over 100 FBR cases were examined, evaluating the effects of variable (1) uranium loading, (2) reflector thickness, (3) fuel bed diameter, (4) frit composition and thickness, (5) control drum composition and design, and (6) reflector temperature.

Figure 11 shows k_{eff} as a function of uranium loading and thickness of the outer beryllium reflector. (The inner reflector and frit are graphite, the outer frit is zirconium, and the fuel bed diameter is 50 cm.) Neutron leakage from the outer reflector strongly affects criticality. Optimum beryllium reflector thickness is about 20-30 cm; much thicker reflectors only marginally reduce critical mass, while much thinner reflectors lose too many neutrons, resulting in excessively high critical masses.

Four moderator/reflector materials have been investigated--Be, ZrH_2, ZrD_2, and C. Beryllium is the best, yielding the lowest critical masses and smallest reactors. Zirconium-hydride-moderated FBRs could not go critical (the highest k_{eff} was slightly less than 1.0), because the external moderator/reflector absorbed too many neutrons. Externally moderated reactors require a small T/L^2 (age divided by thermal diffusion length). Hydrogenous moderators have a large T/L^2 (approximately 5), and many thermal neutrons never return to the fuel. If the fuel particles are dispersed throughout the moderator, hydrogenous moderators are practical. Finally, zirconium deuteride and carbon are practical but result in higher critical masses. Also, carbon-moderated reactors are larger than beryllium-moderated reactors. A carbon moderator/reflector must be approximately twice as thick as the equivalent beryllium moderator/reflector.

The effect of fuel bed diameter is shown in Figure 12. Uranium loading for a given k_{eff} decreases slightly with increasing diameter. This results from a more favorable geometry as diameter increases. If thickness of the outer moderator/reflector is held constant, the fraction of neutrons that return to the fuel bed increases with fuel bed diameter. However, critical mass eventually increases with diameter, since this geometric effect will not be as important as the increasing fuel bed area.

The FBR frits cannot absorb too many of the returning neutrons. Zirconium makes an attractive outer (cool) frit but cannot operate in high-temerature H_2. High-temperature Hastelloy or Inconel frits are practical if thickness does not exeed approximately 1 cm. Neutron absorption becomes excessive for thicker frits. Ceramic (ZrC) or graphite high-temperature frits are also acceptable.

Radial peak to average power ratios are about 2/1 for the FBR, with minimum power density near the center of the fuel bed. This variable power density can be readily handled. The large heat transfer area in the bed makes ΔT between the particles and the coolant very small, even in the peak power region.

In summary, the FBR has reasonable critical masses, of the order of 10-40 kg ^{235}U, and it does not appear to have any serious neutronic problem. The external moderator must have a low absorption cross section and be thick enough to prevent excessive neutron leakage. The frits must not strongly absorb neutrons that diffuse back into the fuel.

Thermal-hydraulics performance of the FBR can be accurately predicted using correlations for pressure drop and heat transfer coefficient in packed beds. For pressure drop in packed beds, the correlation of Ergun (Eckert and Drake, 1972) is used.

Pressure drop across the frits is taken to be 40 percent of the total pressure drop across the bed and frits. This helps keep gas flow uniform even if local bed voidage varies. Frit thickness and porosity can be adjusted to compensate for axial power variations.

Analysis of axial power variations has not been carried out, since analyses have been one-dimensional. However, such variations should be important only near the ends of the fuel bed. Differences between the maximum and mixed-mean coolant outlet temperatures are expected to be small.

Temperature and pressure drop distributions in FBR packed beds with radially nonuniform power densities are determined by a computer code. Beds with nonuniform power density have essentially the same pressure drop as beds with uniform power density, when normalized to equal average powers.

The thermal-hydraulic behavior of the FBRs has been examined for a large number of cases as a function of (1) fuel bed diameter and thickness, (2) inlet and outlet temperature, (3) power level, (4) fuel particle diameter, and (5) inlet pressure.

Table 1 compares thermal hydraulic parameters for three representative point designs. Case 3 has the same fuel bed diameter as Case 2 (75 cm) but a higher power (250 MW versus 100 MW) and lower pressure drop (25 psi versus 36 psi). The higher power and lower pressure drop in Case 3 are achieved with split flow. The fuel bed is radially split into two parts, with the inlet plenum between the outer and inner halves. Coolant flows radially through each half to adjacent outlet plenums. Coolant flows in from both ends of the reactor and out from both ends. The shorter flow path and reduced mass flow through each half greatly increase power output for the split flow design. Coolant ducting is more complex, however.

Maximum power for He-cooled FBRs is about 300 MW(t). A single reactor could generate up to about 100 MW(e) in a closed-cycle cw power system. At this power, reactor mass is much less than that of the rest of the system, so that multiple reactors could be used if higher power outputs are desired. However, it probably will be decades before cw powers above 100 MW(e) are needed in orbit.

Maximum power is much higher for open-cycle FBRs because of the higher specific heat, lower density, and higher ΔT across the core for H_2 coolant. Hydrogen-cooled FBRs will deliver about 5 times as much thermal power as He-cooled FBRs. Conversion efficiencies

Table 1 Fixed-Bed Reactor Point Designs

	Case 1	Case 2	Case 3 (Split-Flow)
Thermal power (MW)	15	100	250
Inlet temperature (^{o}C)	200	200	200
Outlet temperature (^{o}C)	1000	1000	1000
Inlet gas pressure (atm)	100	100	140
Fuel bed diameter (cm)	30	75	75
Fuel bed thickness (cm)	3	6	6
Fuel particle diameter (μm)	400	500	500
Pressure drop (frits and bed) (psi)	21.5	35.9	24.7
Temperature difference (fuel surface-gas (^{o}C))	10	7	16
Fuel surface heat flux (W/cm^2)	20.1	13.1	34.4

(thermal to electric) for high-temperature open-cycle FBRs will be greater than for closed-cycle FBRs, in the range of 40-50 percent for turbine power systems. The capability for high efficiency and high power will allow open-cycle FBRs to generate hundreds of megawatts (electric). This should meet any power requirement for a long time to come.

Scaling of FBRs has been examined from two perspectives:

1. How does size (and mass) of optimum FBRs scale with power level?
2. What is the size (and mass) of the smallest number of FBR designs that can comfortably span the expected power range of applications?

The first perspective assumes that time and resources are sufficient to develop and optimize reactors for each particular mission power. The more realistic second perspective recognizes that even if the same reactor technology is used, a great deal of engineering and testing is required to develop an operational system at a given power. Since the reactor is a small part of the system mass, it appears practical to use two or three nonoptimal reactors to cover the power range, without a major penalty in system weight. Any small penalty due to extra reactor weight would be offset by the large savings in development.

Assuming the first perspective, Figure 13 shows fuel bed diameter for "optimized" He-cooled FBRs as a function of power. The optimum case is taken to be an FBR with the following characteristics: He

pressure, 100 atm; fuel particle diameter, 500 μm; fuel bed thickness, 6 cm; ΔT across the core, 800°C (200°-1000°C); and pressure drop (bed plus frits), 3 atm.

While these conditions are probably not truly optimum, they are reasonably representative. The effects of changing operating parameters from the optimum (canonical) case are also shown.

Figure 14 shows the comparable plot for open-cycle H_2-cooled FBRs with the same design characteristics. Behavior is similar to the He-cooled reactors, except that power output is approximately 5 times greater for a given fuel bed diameter.

Figure 15 shows how the reactor mass scales with FBR power for the design conditions shown in Figures 13 and 14. The outer moderator is more than one-half of the total reactor mass.

Assuming the second perspective, the entire power range can be handled by two FBR sizes: (1) a 50-cm-diameter fuel bed for low powers and (2) an 80-cm-diameter fuel bed for high powers.

Figure 16 shows pressure drop as a function of power for both He and H_2 coolants. The boundary between low and high power appears to be about 50-100 MW(t) for He-cooled reactors and about 200-400 MW(t) for H_2-cooled reactors. The reactor will be somewhat larger than optimum at the low end of its power range and somewhat smaller than optimum at the high end.

There appear to be no major penalties to using two FBR designs to cover the expected power range. Each design has reasonable operating parameters (pressure level, pressure drop, and ΔT (fuel particle to gas)). The FBR appears simple to construct and operate (Figure 8). The main components are fuel particles, inner and outer frits, inner moderator, outer moderator, coolant plenums, control drum, coolant ducts, and pressure vessel.

FBR fuel particles would be fabricated as HTGR fuel particles are now. If they operate in high-temperature H_2, they would be coated with ZrC to minimize attack. Particles would fill the annular space between the inner and outer frits, forming a complete fuel assembly. The assembly could be vibrated during filling to ensure uniform packing.

The outer frit would be metal. In the closed-cycle mode, temperature is about 500-600 K; in the open-cycle H_2-cooled mode, only about 200 K. A zirconium frit is desirable for neutronics. It does not react with He but can react with H_2 at elevated temperatures. However, the above temperature is low enough that no reaction should occur. The outer frit would be wound from wire and sintered to form a rigid cylinder. The large frits tested in RBR experiments were strong and distortion free, and had low pressure drop.

The inner frit operates at the coolant outlet temperature. For FBRs operating only with He coolant, refractory metal frits can be used, since turbine materials limit the outlet temperature to about 1500 K. For open-cycle FBRs, the inner frit will be exposed to hot H_2, and refractory metal frits are probably not suitable. However,

Hastelloy or Inconel can operate in high-temperature H_2 up to about 1400 K.

For H_2 outlet temperatures above about 1400 K, the inner frit will probably be nonmetallic porous ceramic. Single ceramic frits would be difficult to fabricate and could be cracked by thermal and/or mechanical stress. A more promising approach is to build up a frit from segmented overlapping plates, which would be individually supported by the inner moderator. The outlet coolant would flow radially through the frit and then through axial channels to the exit duct at the end of the reactor.

The graphite inner moderator would run hot and have no structural stress. For H_2-cooled FBRs, the graphite would be coated with ZrC or NbC. Some cooling of the inner moderator would be necessary to remove energy deposited by neutron and gamma heating.

An inner moderator constructed from plates appears more reliable and less subject to thermal stress than a monolithic cylinder. The plates would be wedge shaped and assembled into an annulus with a central channel for the hot exit coolant. Coolant passages would be provided by grooving one face of each plate. The plates would be coated with ZrC or NbC for protection against H_2.

The outer moderator will probably be Be, though ZrD_2 and graphite could be used. A plate assembly appears desirable, with surface grooving for coolant channels. About 10-20 percent of the inlet flow will be required to cool the outer moderator, depending on design. Coolant flow would split at the reactor inlet, with the outer moderator portion distributed by a plenum between the moderator and pressure vessel. Most of the flow would enter the main coolant plenum next to the fuel bed. Local flow rate through the bed would be controlled by varying the thickness and/or porosity of the inner and outer frits in order to offset power gradients. Coolant from the outer moderator mixes with the main flow before entering the fuel bed. Rotatable control drums are preferable to axial control rods, since they will not cause axial power peaking. Azimuthal power asymmetries should be small. The control drums would be cooled by flow circuits in parallel with the rest of the reflector.

The inlet and outlet coolant ducts through the end reflector and shield would be curved for minimum neutron streaming. Streaming is important for manned systems, because it increases neutron dose rate outside the reactor. For unmanned systems, streaming effects are not expected to be very important. Sensitive components (i.e., electronics) would be located behind a shadow shield, at the opposite end from the coolant ducts. Curving the coolant ducts appears practical but has not been investigated in detail.

The pressure vessel would be made of high-strength titanium or steel. Pressure vessel thickness will depend on the operating pressure and reactor size; typically, it will be in the range of 1-2 cm for an operating stress of approximately 50,000 psi.

THE RBR (ROTATING-BED REACTOR)

Neutronic analyses for the RBR were carried out with the same techniques used for the FBR. End effects are expected to be greater for the RBR than for the FBR, since the former has an exit nozzle and no inner moderator/reflector. Critical loading for finite-length RBRs is taken to be the same as the infinite cylinder at $k_{eff} = 1.2$, computed with the one-dimensional ANISN model.

Almost 100 RBR cases were analyzed, investigating the effects of (1) reflector composition and thickness, (2) fuel bed thickness, (3) degree of fluidization, (4) reactor size, (5) H_2 pressure, (6) frit composition and thickness, and (7) moderator temperature.

RBRs have larger critical masses than FBRs, because there is no internal moderator/reflector. Figure 17 compares k_{eff} for Be-moderated RBR and FBR reactors for different uranium loadings. At a given k_{eff}, uranium loading is a factor of 2-3 higher for RBRs. Reflector thickness strongly affects critical loading in the RBR. Beryllium thickness less than about 20 cm makes neutron leakage excessive; thickness substantially more than about 30 cm saves relatively few neutrons and reduces critical mass by a small amount.

The effect of fuel bed thickness on RBR criticality is small (Figure 18). Changing bed thickness from 5 to 11 cm changes k_{eff} by about 3 percent. This insensitivity to thickness is a result of the "blackness" of the reactor core. Almost all thermalized neutrons that diffuse back from the reflector are absorbed in the fuel bed, and few survive to enter the cavity. The high peak-to-average power ratios for the RBR (about 3/1) are caused by the blackness of the fuel bed.

The fuel bed blackness also means that bed expansion has virtually no effect on k_{eff}. Figure 19 shows no change in k_{eff} in the range from fully settled (solid fraction of about 0.65) to fully expanded (solid fraction of about 0.3). Having RBR criticality independent of bed fluidization eliminates concerns about bed movements initiating power excursions.

Figure 20 shows that k_{eff} is insensitive to reactor size over the range studied. This behavior is also seen in the FBR. Figure 21 shows how k_{eff} depends on H_2 pressure. Nominal pressure (P_0) is 70 atm. Reactivity drops slightly (by about 1 percent) as P/P_0 decreases from 2.0 to 0.5. The small magnitude and favorable sign of the effect make H_2 depressurization not a safety concern.

Neutron absorption in the frit significantly affects RBR criticality. Figure 22 illustrates how k_{eff} changes with equivalent thickness of a zirconium frit for an RBR of fixed size and loading. The Zr frit is assumed constant in actual thickness (1 cm), but its density factor varies from $\rho = 0$ to $\rho = 8$ ($\rho = 1$ is nominal solid density), and k_{eff} decreases by about 0.25 over this range.

Strongly absorbing frits, such as stainless steel or titanium, will seriously degrade k_{eff}. They are only practical for thicknesses of approximately 1 cm or less.

The frit absorbs neutrons returning from the outside moderator/reflector. If absorption is too great, the reactor cannot go critical. Because the frit absorbs neutrons before they reach the uranium fuel, its effect is much greater than if it were mixed with the fuel.

Part of the outer moderator could be rotated as a frit to contain the fuel. This is a promising approach, particularly with beryllium, since it is strong and light. Structural stress on the rotating structure would be smaller, and frit absorption would be eliminated. Future studies should examine this option.

RBR thermal-hydraulics have been investigated experimentally and analytically. Experimental work is discussed later. Previous analyses of rotating fluidized beds did not allow for temperature rise through the bed and, instead, assumed a uniform temperature. Self-consistent numerical models have been developed that incorporate volume heating of the bed and radially variable power densities.

Figure 23 illustrates how bed voidage varies with radial position and rotation speed for a 1,000-MW RBR of constant size ($L = R = 0.5$ m). Fuel loading is equivalent to a settled-bed thickness of 5 cm. The bed is fully settled above 1,025 rpm. As speed drops below this value, the inner edge of the bed begins to fluidize. The fluidized region becomes thicker as speed decreases further. The outer part of the bed remains settled until approximately 350 rpm is reached. At this point, the outer edge begins to fluidize. As speed decreases further, the fluidized bed expands. The fluidized bed becomes twice as thick as the original settled bed at 175 rpm. Voidage (i.e., coolant volume) at the inner boundary is then 80 percent. Average voidage is less, i.e., 67 percent, while voidage at the frit is 45 percent.

These analyses do not allow for the effects of bed mixing on temperature and voidage distributions. Rapid bed mixing would make the temperature and voidage more uniform. Experimental evidence suggests such behavior. Gas temperature increases rapidly across the outer settled layer of particles and then becomes essentially uniform in the fluidized region. However, the experiments have not operated at conditions comparable to the RBR's, and the importance of this mixing process in the RBR is not certain.

The RBR parameter space is much wider than that of the FBR, since it has more operating variables. It is thus more difficult to define optimum designs. The major variables affecting performance are (1) degree of bed expansion, (2) settled-bed thickness, (3) bed diameter, (4) base pressure of coolant, (5) particle diameter, and (6) inlet and outlet temperature.

Figure 24 shows RBR rotation speed versus power level and degree of bed expansion (H_{max}/H_0). The pressure drop curves (dashed lines) show ΔP (atm) across the bed. Bed diameter is 50 cm, and thickness is 4 cm. At a given power, pressure drop is reduced by increasing bed expansion. However, expansion limits are set by particle carry-over at the inner boundary of the fluidized bed. A limit of $H_{max}/H_0 = 2$

is assumed. Carry-over can start if expansion is much greater than this limit.

Figure 25 shows reactor power as a function of radius for $H_{max}/H_O = 2$, and $H_O/R = 0.1$. (H_O is settled-bed thickness). The solid curves are lines of constant P and range from 1 to 10 atm. The dashed curves are lines of constant rotational speed (rpm). The RBR can deliver up to about 5,000 MW(t) over the size range considered at reasonable ΔP's (5-10 atm). Rotation speeds are also reasonable and range up to a maximum of about 1,000 rpm.

Table 2 shows three RBR designs of different size and power rating. Values of "fuel" mass in Table 2 and the figures refer to total particle mass (UC + ZrC), not uranium mass. The weight fraction of uranium is the mass of uranium divided by total fuel mass. In small RBRs, this can range up to about 50 percent; for larger RBRs it will be 10 percent or less.

The RBR and FBR have comparable mass for the same fuel bed diameter and range from about 1 to about 3 tonnes (t), depending on size. The dependence of RBR mass on power will be similar to that of the FBR, except that power rating will be about tenfold greater.

The mechanical design of the RBR has been examined. The RBR frit would be supported by a shaft at the end opposite the exit nozzle. The shaft would be radially centered and positioned by a gas bearing, with a separate gas bearing to counter end thrust. Clearances would be of the order of 1 mil. The assembly would be driven by the inlet coolant acting on a set of small blades on the shaft.

End reflectors would be incorporated in the rotating frit assembly. They would be cooled by a small flow of gas passing from the inlet plenum through the end reflectors into the cavity. A labyrinth seal between the rotating assembly and the nozzle end of the reactor would limit gas leakage from the coolant plenum.

The shaft of the rotating RBR assembly would contain a channel to admit fuel particles. After the reactor is in orbit, the assembly would be spun up to operational speed. Particles would be blown into the cavity and caught by the walls of the rotating frit.

The RBR can unload fuel as well as load it in a few seconds. This would have important safety and operating advantages. The same channel would be used for both unloading and loading, with a movable plug to block the exit nozzle when unloading. Experiments with RBR simulations should be carried out to confirm this capability.

In summary, the RBR is neutronically similar to the FBR, except that its uranium loading is a factor of about 2 greater at equivalent k_{eff}. This is a result of not having an internal reflector. The mobile fuel bed should not cause any safety problems. Thermal-hydraulics of the RBR are much more complex than those of the FBR. Power outputs for the RBR are much higher than those for the FBR for the same reactor size and pressure drop. Power levels of several gigawatts can be achieved with RBRs about 1 m in diameter. RBRs are more complex mechanically than FBRs owing to the need for bearings and

TABLE 2 Rotating-Bed Reactor Point Designs

	Case 1	Case 2	Case 3
Thermal power (GW)	0.25	1	5
Radius (cm)	15	25	40
Length (cm)	15	50	80
Fuel particle diameter (μm)	500	500	500
Hydrogen flow (kg/s)	5.7	22.8	114
Inlet temperature	300	300	300
Outlet temperature	3000	3000	3000
Cavity pressure (bar)	100	100	100
Fuel bed thickness (cm)	3	2.5	4
Fuel mass (kg)	42.2	206	800
Pressure drop (bar)	3	5	15
Bed expansion (average) (%)	100	50	100
Fuel surface heat flux (kW/m^2)	4,200	3,440	4,430

labyrinth seals. However, the RBR appears to be constructable without undue difficulty.

POWER SYSTEM PERFORMANCE

The characteristics and performance of space electric systems using FBRs and RBRs have been investigated. For the RBR, open-cycle, turbines, and MHD cycles were considered; for the FBR, both closed-cycle and open-cycle turbine cycles were examined.

The size, weight, and layout of the various components making up a complete power system were estimated as a function of power, operating time (for open cycle), coolant temperature (both inlet and outlet), generator efficiency, etc. These components were then integrated into self-consistent designs.

Figure 26 depicts a 100-MW(e) closed-cycle system using a He-cooled FBR (left front of the drawing). Behind the FBR are the in-line turbine, compressor, and superconducting generator. The vessels above and below the power generation equipment are helium to lithium heat exchangers. Reject heat from the helium coolant is transferred to liquid lithium and then radiated to space using the liquid-droplet radiator (LDR, proposed by Hertzberg. The droplet spray nozzles and collectors are not shown but are mounted on the Astromast structure that holds the heat exchangers.

RBR-powered MHD generators were investigated. Initially, MHD was viewed as the most promising route for high-power, open-cycle operation. This assessment now appears to be wrong. Outlet

temperatures of at least 2750 K are needed to operate a H_2 MHD
generator. At this temperature, however, although MHD generators are
feasible, performance is poor. Higher temperatures, e.g., 3000 K,
result in more practical generators, but efficiency is still only
about 20 percent (H_2 coolant enthalpy converted to electricity).
This low efficiency makes H_2 mass throughput very high and greatly
increases H_2 storage requirements.

High-temperature turbines are much more attractive than MHD
generators. Turbines with H_2-cooled blading were operated
successfully by General Electric (GE) in the early 1960s at turbine
inlet gas temperatures of about 2500 K. Metal blade temperature were
only about 700 K. For these cycles, turbines can extract 45-50
percent of the H_2 enthalpy. Thus although turbines would operate at
lower reactor outlet temperatures, turbines can generate twice as much
electrical energy per kilogram of H_2 coolant as can MHD generators.

Advanced turbines using high-temperature ceramic c. carbon-carbon
blading are also a potential option for high-power, high-efficiency,
open-cycle systems. The relatively short operating life required for
these systems will permit the use of blade materials not practical for
commercial turbines. Extensive work has been done on both types of
blades, most recently on carbon-carbon blades.

Lightweight turbines and generators are being developed for
airborne applications. A preliminary version of a 20-MW(e)
superconducting generator has been tested by GE and the Air Force
Aero-Propulsion Laboratory (APL). This device weights only about 1 t.
Single units can be scaled to higher powers, probably about 50 MW(e).
Multiple generators could be used for powers above 50 MW(e). Small
generators, i.e., up to a few megawatts (electric) in output, would
use Sm-Co permanent magnets for field excitation. A 5-MW(e) permanent
magnet generator is under construction for APL. Scaling relations for
turbines, compressors, and generators for this study were based on
work by APL, Rocketdyne, and Garrett.

Closed-cycle Brayton-cycle designs using He-cooled FBRs were
developed for a range of cw powers from 100 kW(e) to 100 MW(e). Both
regenerative and nonregenerative cycles were investigated. Table 3
lists component sizes and weights for four power system sizes: 5, 20,
50, and 100 MW(e). (The shield weight is a full 4π shield for manned
space systems. Unmanned systems would use a much lighter shadow
shield.) At high power levels, heat exchangers (reject H and the
recuperator) dominate. Radiator weights are not included.

Figure 27 shows the main component weights (excluding shield) for
He-cooled closed-cycle FBRs as a function of output power. Lithium
liquid-droplet radiators (LDR) would be used for heat rejection. LDRs
are projected to be about an order of magnitude lighter than
conventional fin-tube or heat pipe radiators. Even with LDRs,
however, the radiator is still the heaviest part of the system.

With LDRs, it appears feasible to construct cw power systems to
deliver several tens of megawatts (electric). If conventional

Table 3 Regenerative Brayton Cycle

	Output Power MW(e)			
	5	20	50	100
Reactor weight[a](t)	0.70	1.87	3.57	7.20
Turbine weight(t)	0.16	0..47	1.04	2.08
Turbine diameter (m)	0.4	0.6	0.8	2 x 0.8
Compressor weight (t)	0.32	0.94	2.08	4.16
Alternator weight (t)	0.23	0.91	2.27	4.55
Recuperator weight (t)	0.3	1.2	3.0	6.0
Recuperator volume(m^3)	0.5	2.0	5.0	10.0
Reject HX weight (t)	0.44	1.76	4.40	8.80
Shielding weight (1 rem/h contact) (t)	20.0	25.0	25.0	25.0
Total weight (no shield or radiator) (t)	2.14	7.15	16.36	32.79

[a]T_{cav} = 1500 K; T_{re} = 200 K.

radiators have to be used, weight limitations will restrict cw power levels to a few megawatts (electric).

Figure 28 illustrates the trade-offs between closed- and open-cycle power systems using FBR reactors. The break-even point is the integrated operating time for which open- and closed-cycle systems have equal mass (including the stored H_2 and tankage in open-cycle systems). The break-even point appears to be about 6,000 s and is relatively independent of system power output. For times less than 6,000 s, open-cycle systems are lighter; for times greater than 6,000 s, closed-cycle systems are lighter.

High-power applications being considered probably would not require operating times exceeding about 6,000 s. Open-cycle systems are thus likely to be preferred over closed-cycle systems for high powers.

At the low end of the power spectrum (hundreds of kilowatts (electric) to about 20 MW(e)), total weight is low enough that practical cw systems can be carried in a single shuttle load.

In summary, turbogenerator power systems appear to be the most promising approach for both FBRs and RBRs. MHD generators require very high operating temperatures, and conversion efficiency is considerably lower than is the case for a turbine system. The FBR can be used in both closed-cyle (He-cooled) and open-cycle (H$_2$-cooled) systems. Open-cycle generation up to several hundred megawatts (electric) appears possible with the FBR, using H$_2$-cooled turbines and superconducting generators. High powers, up to about 2 GW(e), appear possible with the RBR. Closed-cycle FBRs will require liquid-droplet radiators (LDRs) for heat rejection if power levels of more than 1-2 MW(e) are desired. Continuous power outputs up to about 100 MW(e) appear feasible with the FBR. The break-even point between closed- and open-cycle systems is an integrated operating time of approximately 5,000-6,000 s. For longer operating times, closed systems will be lighter; for shorter operating times, open-cycle systems will be lighter.

ENVIRONMENT AND SAFETY

The RBR and FBR can meet the safety design requirements formulated for the SP-100 project. These are as follows:

1. A launch accident or abort cannot cause criticality.
2. The reactor must remain subcritical if immersed in water or other fluids.
3. A negative temperature coefficient of reactivity must be achieved.
4. The reactor will not operate until a stable orbit is achieved.
5. The reactor must have reboost capability if operated in low orbit (with less than a 300-year decay time).
6. There must be two independent safety shutdown systems not subject to common mode failure.
7. There must be an independent shutdown heat removal system or independent heat removal paths.
8. The unirradiated fuel should pose no environmental hazard (i.e., ^{235}U should be used).

The FBR would be loaded prior to launch and designed to withstand launch accidents and aborts. External hydrogenous moderators cannot make the FBR critical, so that immersion in water or propellants will not cause criticality accidents. At the projected uranium loadings, an external hydrogenous moderator will have a maximum k$_{eff}$ of about 0.9.

The RBR would be loaded after achieving a stable orbit. Fuel would be blown into the cavity after frit rotation was established.

The FBR and RBR should be easier to control than commercial power reactors. The external neutron moderation results in long neutron lifetimes and makes reactivity sensitive to poisons in the reflector. Both effects aid control.

Figure 29 shows the rate of power rise for a RBR subjected to a step increase in reactivity. An insertion of 20 dollars above prompt critical is assumed. Even for this very large insertion, power e-folding time is relatively long (30 ms) and within the control capability of the reactor. Power e-folding time for light water reactors (LWRs) would be almost 2 orders of magnitude faster for a comparable reactivity insertion. FBR behavior should be similar to that of the RBR. FBRs and RBRs are expected to be safer than conventional reactors in terms of criticality control.

Temperature coefficient of reactivity was examined for one FBR design. In this design, the coefficient was slightly positive, with an increase in k_{eff} of about 0.25 percent per 100 K rise in reflector temperature. The positive coefficient can be easily managed, since the corresponding power e-folding time is long (seconds) and a compensating reactivity adjustment can be easily made. Negative temperature coefficients are more important if neutron lifetime is short, which is the case for fast reactors or LWRs. It appears likely that the FBR design can be adjusted to achieve negative temperature coefficients. One possibility is to incorporate some non-(1/V) poison in the reflector, which would compensate for the effect of temperature on reflector absorption cross section.

RBRs and FBRs would not operate until the spacecraft is in a stable orbit. If they operated in low orbit (with less than a 300-year life), they would be boosted into a high orbit at the end of life so that fission products would decay in space.

RBRs and FBRs would have two independent safety shutdown systems not subject to common mode failure. The FBR would have two sets of safety rods in the outer and inner reflector, either of which could shut down the reactor. The RBR would have a set of safety rods in the outer reflector, plus the capability for injection of neutron poison into the cavity (e.g, B_4C particles).

FBRs and RBRs would have independent shutdown heat removal systems. For the RBR, this probably would be a separate pressurized H_2 gas circuit. H_2 would flow through the core to remove afterheat, with flow rate controlled by valves. The integrated amount of coolant required to remove afterheat is only a few percent of the main coolant inventory, since the afterheat energy is a few percent of total fission energy. Multiple H_2 storage tanks and valves could be used for reliability.

Afterheat in a RBR will quickly decay to very low levels. For example, after operation for 100 s at 100 MW(t), the afterheat power drops to about 100 W a day after shutdown. Long-lived residual radioactivity in RBRs will also be very low. For an integrated power equivalent to a 1,000-s burn at 500 MW(t) (a representative mission lifetime), approximately 10 Ci (curies) each of ^{90}Sr and ^{137}Cs are

produced. Long-lived radioactive inventories are almost a millionfold smaller than in commercial power reactors.

This low inventory, plus the very large dilution in space if it is dispersed, makes accident consequences very small for RBRs or pure open-cycle FBRs. Principal concern would relate to loss of function, not environmental or safety hazards.

FBRs operating in the cw mode will have larger inventories of fission products and greater afterheat cooling requirements. A 1-MW(e) FBR that operates for several years will have a radioactive inventory about one-thousandth that of a commercial LWR. Potential accident consequences are much less than those for commercial plants, because of both the lower inventory and the greater isolation of the reactor. The cw FBR will have a separate shutdown cooling system, with independent (and redundant) gas circulators, heat exchangers, and radiators. Since the shutdown heat level is only a few percent of the main power load, the additional components do not seriously increase system weight. Leaking pipes and components can be shut off while still maintaining aftercooling. However, coolant leaks from the reactor vessel will result in loss of coolant and eventual failure of the core. Unfortunately, there is no "sump supply" in space.

The FBR and RBR will both use ^{235}U fuel, which poses no environmental hazard. The reactors would be designed to burn up and disperse their fuel during reentry. The particulate nature of the fuel makes fuel unloading a potential option for the RBR and FBR. Designs for unloading RBR fuel have been investigated, and it appears feasible to also unload FBR fuel. This would have important safety benefits, since the fuel could be unloaded into a passively radiating, shielded cask if a loss of coolant situation developed. Unload times of a few seconds appear practical. This would prevent fuel melting and consequent dispersion of fission products.

The ability to unload fuel is also of great benefit for (1) personnel access, (2) maintaining reactivity after extended burnup, and (3) waste disposal. Unloading fuel would greatly reduce the radiation dose from the reactor and could permit direct personnel access (probably with some limited shielding). If long burnup had degraded fuel reactivity to the point where criticality could not be maintained, the spent fuel in the FBR could be unloaded and replaced with fresh fuel. The spent fuel could be unloaded into a small, shielded, passively cooled cask and boosted into an earth escape or high orbit for disposal.

RBR/FBR EXPERIMENTS

Scoping experiments on simulated RBR and FBR particle beds have demonstrated capabilities comparable to those expected for operational reactors. However, these preliminary experiments have not fully explored all conditions appropriate to FBRs and RBRs.

Electrically heated FBR particle beds have operated at power densities of about 1 kW/cm^3. Direct resistance heating and induction heating have both been used, with helium cooling. The beds have been composed of 500-μm BISO particles (used in the HTGR) as well as 1,000-μm stainless steel spheres.

The experimental power density, approximately 1 kW/cm^3, is comparable to the design power in full-scale cw FBRs. Results are very impressive, considering that the average He coolant pressure in the bed was only about 1.5 atm. Higher coolant pressures should allow much higher power densitites. Experiments will be carried out at higher pressures (about 15 atm) in the near future. The measured pressure drops matched predicted values.

Helium outlet temperatures of approximately 1200°C were achieved with BISO-type particle beds. This matches desired FBR outlet t   rature in the closed-cycle mode. Stainless steel frits showed no reaction at 1200°C during the experiments (over about 5 hours). Zirconium frits did react with fuel particles at this temperature, however.   Particle beds have been rapidly ramped up to full power (typically in 2-3 s) without damage to the particles. Bed power has also been rapidly cycled without observable damage to the particles.

All experiments have so far been on He-cooled particle beds. Experiments are planned on H$_2$-cooled beds.

Turning to the RBR-related experiments, an extensive series of hydraulic tests was carried out at Brookhaven in the early 1970s. In these experiments, the hydraulic behavior of rotating fluidized beds was studied under RBR-like conditions, using N$_2$ coolant and glass or copper particle beds as stand-ins for the UC-ZrC fuel bed. The beds were not heated.

The simulated RBR beds were 25 cm in diameter and operated at 10 atm. Beds operated without problems in all three modes--settled, partly fluidized, and fully fluidized. Figure 30 shows a high-speed photograph of part of a rotating, fully fluidized bed. The bed has expanded by about 30 percent in volume from the initially settled state. The view in Figure 30 is along the axis of the rotating bed and shows the inner boundary to be smooth and curved like a liquid meniscus. Particles do not leave the surface of the bed.

High-speed movies were also taken of the rotating beds. In one set of experiments, a layer of colored beads was placed on the inner surface of a rotating settled bed, which was then allowed to fluidize. The colored beads gradually mixed with the rest of the bed. Particle motions were gentle, and no large-scale mixing was observed.

When the bed was taken into the highly expanded state, large-scale bubbling could be observed. Individual particles would break free from the bed surface. Their trajectories would bring them back to the bed, but impact velocities were higher than in the gently fluidized state. This raises the possibility of attrition. This bubbling state always could be brought back to laminarlike fluidization by increasing

rotation speed at fixed gas flow or by decreasing gas flow at fixed rotation speed.

Figure 31 compares the experimental fluidization curves with predicted values as a function of gas flow and bed loading. Fluidization occurs when the pressure drop becomes constant with flow rate. At this point, ΔP simply equals the effective weight of the suspended particles in the applied g field. Increasing gas flow rate only expands the bed and does not change ΔP. The solid line represents the predicted pressure drop through packed beds as a function of gas flow rate.

The hydraulic behavior of the simulated RBR beds appeared to correlate with analytical modes and confirmed that stable operation could be achieved. Although heated beds could not be examined with the apparatus, experiments with rotating combustion fluidized beds have been carried out by other researchers. These studies indicate that stable operation is practical in volume-heated beds. They also indicate that there is considerable radial mixing that tends to make bed temperature profiles uniform except near the inlet frit. In this region, a sudden temperature jump is observed. This type of behavior is encountered in the RBR. It is not expected to affect performance adversely and, in fact, may aid stability.

FBR/RBR DEVELOPMENT REQUIREMENTS

The FBR does not appear to involve resolution of any go/no-go issues. The materials and components have been extensively investigated in connection with other reactor systems. However, some development of particular components will be required, depending on design, and confirmatory testing will be necessary.

Present TRISO fuel particles are suitable for He-cooled systems up to about 1500 K. Testing of ZrC-coated particles should be carried out for H_2-cooled or higher temperature, He-cooled systems. ZrC-coated fuel has been manufactured and appears to have very attractive properties, but confirmatory testing for FBR application will be required.

Testing of high-temperature frits at FBR conditions will also be required. Refractory metals (e.g., Zr) appear satisfactory for He-cooled FBRs at temperatures up to approximately 1400 K. Hastelloy or Inconel also appear satisfactory at this temperature level and could be used with H_2 coolant. However, long-term confirmatory testing should be carried out to demonstrate adequate performance, particularly in terms of creep, and compatibility with coolant, fuel, and the inner moderator.

For temperatures above about 1400 K, ceramic frits appear necessary, and these will require some development. Graphite is a good possibility for He-cooled FBRs; a refractory carbide (e.g., ZrC) could be used for H_2-cooled systems. A promising construction approach uses segmented overlapping frit plates mounted on the

graphite inner moderator plates. Such an assembly of segmented frit plates should substantially reduce vulnerability to thermal and mechanical stress.

A nonnuclear demonstration of the FBR has been proposed. In this program, termed SPIDER (Space Power Integrated Demonstration Reactor), a simulated FBR would be tested at operating conditions, size, and weight representative of the final reactor system. An overall view of SPIDER is shown in Figure 32. The particle fuel bed would be resistance heated by a 440-V, 10-MW power supply. The bed would be cooled with He; coolant pressure would be in the range of 500-750 psi, and outlet temperature about 1000°C. Hydrogen coolant would also be used to simulate open-cycle operation.

Major parameters for SPIDER are summarized in Table 4. The experiment could operate up to about 40 MW in the He-cooled mode, and about 200 MW in the H_2-cooled mode, if heated nuclearly. Electrical resistance heating inputs of this magnitude are not practical. However, it appears feasible to heat a portion of the bed with the available electrical supply to power densities that would correspond to the full-scale output. This would verify thermal-hydraulic performance over the full operating range.

The first phase of the program would be to test SPIDER as a simulated reactor heat source. After successful completion of the first phase, which would take 2 years, the simulated reactor would be integrated with a power conversion (turbine/compressor) and heat rejection system. Power generation output would be about 3 MW(e). This second phase would test the capability of the complete FBR power system to generate power at the predicted operation conditions, with actual components and weights. The second phase would take an additional 2 years.

Besides serving as a demonstration, SPIDER would provide valuable information that would be used in the design of a ground-based FBR prototype. Tests with SPIDER should confirm the ability to

1. achieve predicted power and power densities
2. verify thermal-hydraulic performance for coolant and fuel bed
3. operate in closed- and open-cycle mode
4. quickly switch from closed to open cycle
5. ramp to full power in a few seconds
6. deliver many pulse power cycles
7. achieve high component and system reliability
8. operate safely in off-normal and shutdown conditions (e.g., loss of main coolant system).

In addition, special studies investigating the ability to remotely and rapidly load and unload the FBR fuel bed would be carried out.

The RBR will require substantially more R&D than the FBR. Fundamental studies on fluidized-bed mixing processes need to be carried out when volume heating in the bed is present. Time-varying temperature and voidage distributions in the bed need to be

Table 4   FBR SPIDER Physical Characteristics

Fuel bed	
Outside diameter (cm)	40
Height (cm)	40
Thickness (cm)	3
Bed volume (l)	14
Particulate weight (66% DF)(kg)	90
Power density during operation	
BNL (full core)(w/cm^3)	215
BNL (1/3 core)(W/cm^3)	645
APL (full core)(W/cm^3)	715
Power generation equivalent (MW(t)	
BNL tests	3
APL tests	10
Nuclear operation capability (1,500 psia)	50
Thermal hydraulics	
Inlet He pressure (psia)	750
Bed pressure drop (BNL)(psia)	0.3
Bed pressure drop (APL)(psia)	1.8
Inlet He temperature (^oC)	30
Outlet He temperature (^oC)	1000
Dimensions	
Outside diameter/height of outer (Be or ZrH_2) moderator (cm)	84/40
Outside diameter of inner radial moderator (graphite) (cm)	30
Weight of outer radial moderator (if Be) (kg)	275
Weight of inner radial moderator (kg)	30
Weight of end moderators (graphite) (kg)	350

established, as well as the effect of volume heating on bed expansion limits.

Fuel particle lifetimes in H_2 need to be determined more precisely at the expected temperature and pressure conditions. The effects of thermal cycling and mechanical attrition in the fluidized bed on particle lifetime should also be examined. Fuel lifetime studies can be carried out using small rotating or 1-g linear beds fluidized with high-pressure, high-temperature H_2. Bed heating could be provided by heating either the H_2 or the particles. Furthermore, fission product loss rates from the fuel at the expected reactor conditions need to be determined. Designing fuel particles to minimize loss rate should also be investigated.

The full-scale cold flow RBR experiments should be extended to examine fuel loading and unloading. In addition, transient start-up

and shutdown processes should be investigated, with particular regard to how rapidly uniform fluidization can be established.

Heat transfer experiments in a full-scale simulated RBR should also be carried out, using cyclic variations in inlet coolant temperature. The phase lag in outlet coolant temperature provides information on particle-to-gas neat transfer coefficients at representative RBR conditions, which can serve to confirm predictions.

SUMMARY AND CONCLUSIONS

Compact high-performance nuclear reactors can be based on direct cooling of HTGR-like nuclear fuel particles. The very large surface area of the particles allows very high-power densities, high coolant outlet temperatures, small temperature differences between fuel and coolant, and very fast start-up and shutdown of the reactor.

Two reactor concepts based on HTGR particles have been investigated in detail. In the first, the Fixed-Bed Reactor (FBR), the particles are packed between two porous frits, forming a relatively thin annular fuel bed through which coolant flows radially. Neutrons are moderated in external reflectors outside and inside the annular fuel bed. The FBR can operate for years in the cw Brayton-cycle mode with inert gas coolant (e.g., He) and a turbogenerator/compressor power system. The FBR can also operate in the open-cycle mode with H_2 coolant, generating very high power outputs for pulsed-energy devices. The same reactor can operate bimodally, switching back and forth between open- and closed-cycle modes when desired.

The FBR can deliver, from a practical device of about 1 m x 1 m in diameter and height, power outputs up to about 300 MW(t) in the He-cooled mode and about 1,000 MW(t) in the H_2-cooled mode. Operating time in the closed-cycle mode can be very long (months to years), depending on power, but is limited to a few hours at most in the open-cycle mode, because of H_2 storage requirements.

The FBR can satisfy almost all of the potential applications with its generation capability. Two basic size reactors can cover the cw power range from approximately 100 kW(e) to approximately 100 MW(e) without serious weight penalty. Open-cycle H_2-cooled generation levels will be approximately 5 times the cw level. Coolant outlet temperature capability is high, about 1500 K in the cw mode and about 2000 K in the open-cycle mode. Power generation efficiency will be high under these conditions, about 30 percent for closed-cycle operation and 40-50 percent for open-cycle operation.

Relatively little development is required for the FBR. Present HTGR fuel particles can be used for closed-cycle operation. Some modification of the particles to use ZrC coatings probably is necessary for open-cycle operation, but development of such particles has already been under investigation for commercial purposes.

The main development task for the FBR appears to be the high-temperature (inner) frit. Refractory metal frits appear

satisfactory for outlet temperatures up to about 1500 K. Refractory ceramic frits appear to be practical for much higher temperatures but must be tested in representative reactor configurations.

For very high power outputs and/or very high coolant outlet temperatures, the FBR would be superseded by the second reactor concept, the Rotating-Bed Reactor (RBR). In the RBR, the annular fuel bed is held inside a rotating porous frit. Coolant passes radially through the frit at low temperature and then through the particle fuel bed and finally exits at high temperature from a nozzle at one end of the RBR internal cavity.

The RBR can only operate in the open-cycle mode with H_2 coolant. The fuel particle bed can operate in either of three states--fully settled, partly fluidized, and fully fluidized. At a given coolant flow rate, the fuel bed state is determined by frit rotation speed. Higher speeds are required for settled-bed operation and lower speeds for fluidized-bed operation.

Output powers up to about 5,000 MW(t) can be generated with relatively small size RBRs, i.e., 1 m x 1 m in length and diameter. RBR coolant outlet temperatures up to the limit of materials capability, approximately 3000 K, can be achieved.

The RBR would be used for direct thrust applications, where high power and high specific impulse ($I_{sp} \sim 1,000$ s) are required, and for generation of very large electric power outputs using MHD generators or high-temperature turbines.

Development requirements are substantially greater for the RBR than for the FBR. The fluid dynamics of volume-heated fluidized beds has not been adequately explored, and fuel particle behavior in very high temperature H_2 needs to be further investigated. Since the integrated operating life for the RBR is at most a few hours because of H_2 mass requirements, long-term material performance under irradiation does not appear to be an issue.

The FBR and the RBR appear to be promising, high-performance compact reactors that are designed to meet a variety of applications and a wide power range. They are based on the small-diameter-particulate fuels developed for the HTGR, for which there is extensive experience and understanding.

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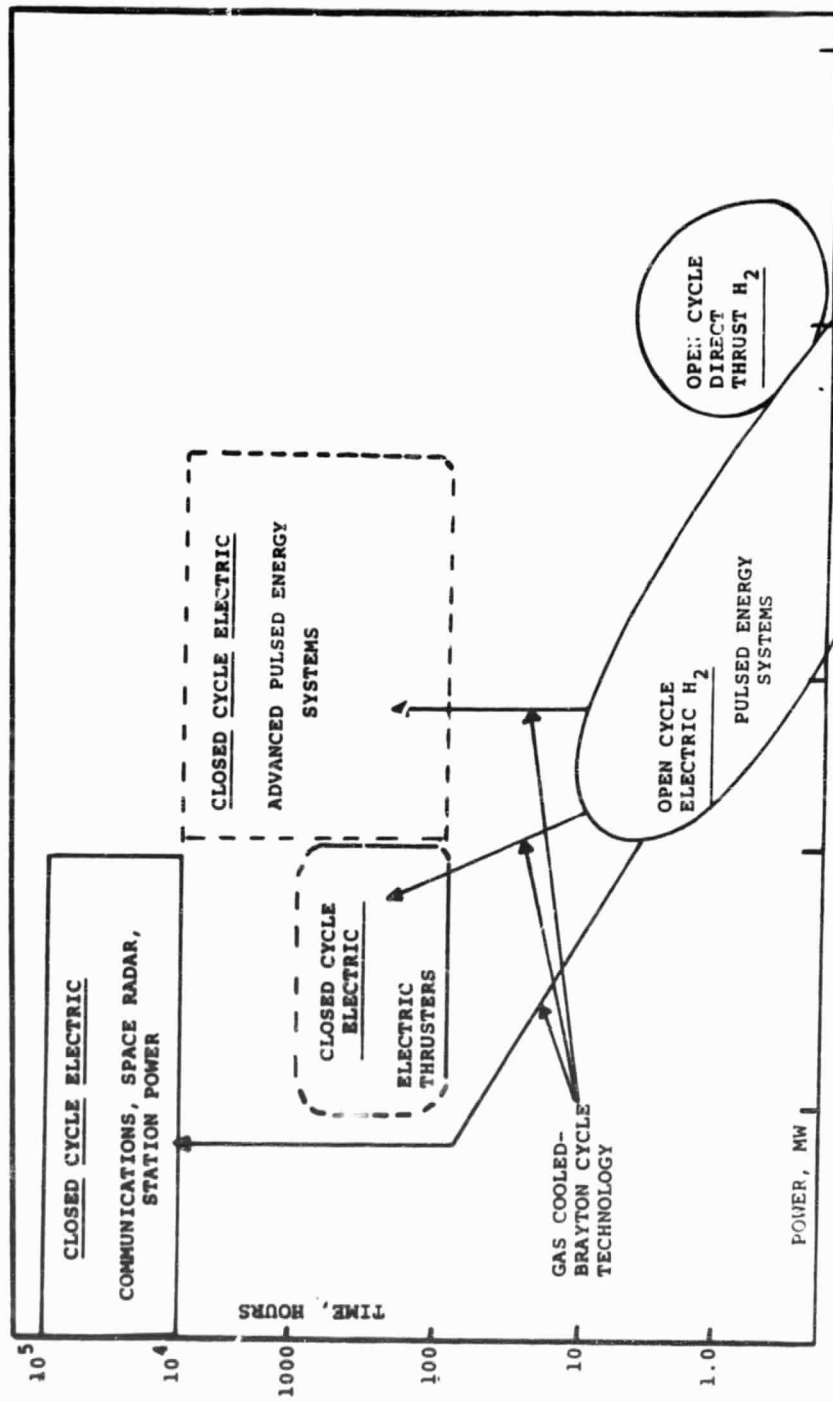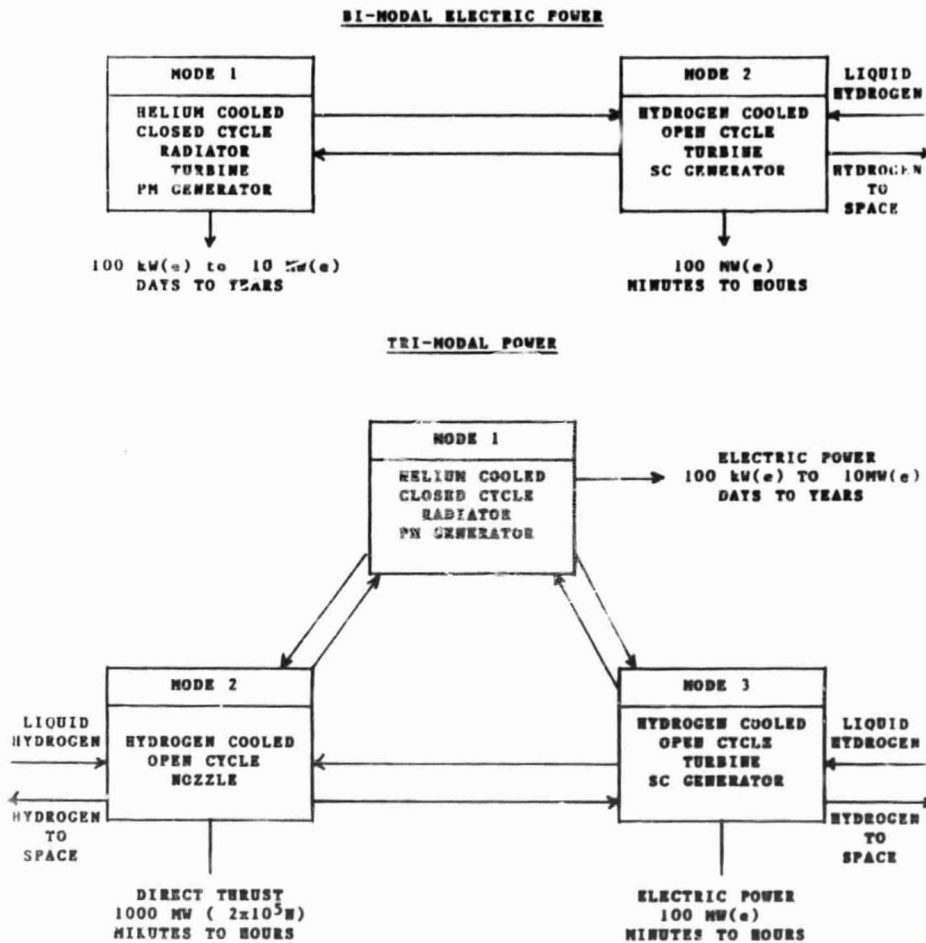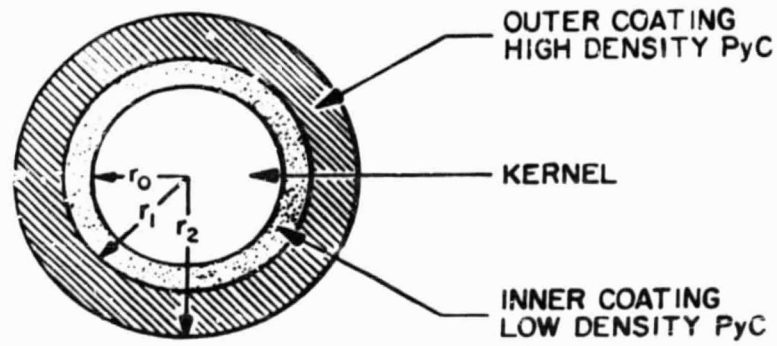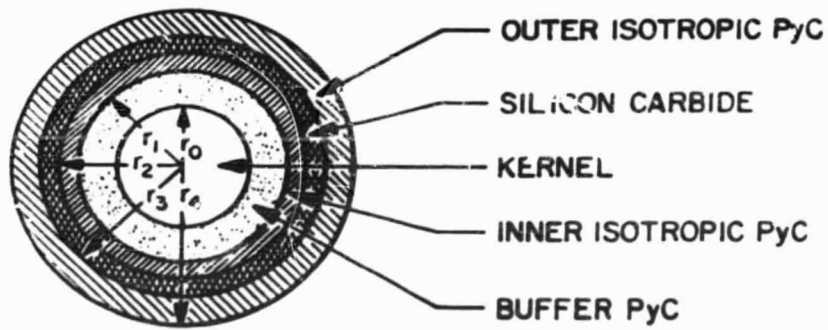
FIGURE 1 Space nuclear power regimes.

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BI-MODAL ELECTRIC POWER

MODE 1	MODE 2	LIQUID HYDROGEN
HELIUM COOLED CLOSED CYCLE RADIATOR TURBINE PM GENERATOR	HYDROGEN COOLED OPEN CYCLE TURBINE SC GENERATOR	HYDROGEN TO SPACE

100 kW(e) to 10 MW(e) DAYS TO YEARS

100 MW(e) MINUTES TO HOURS

TRI-MODAL POWER

MODE 1

HELIUM COOLED CLOSED CYCLE RADIATOR PM GENERATOR

ELECTRIC POWER 100 kW(e) TO 10MW(e) DAYS TO YEARS

MODE 2

LIQUID HYDROGEN

HYDROGEN COOLED OPEN CYCLE NOZZLE

HYDROGEN TO SPACE

DIRECT THRUST 1000 MW (2×10^5 N) MINUTES TO HOURS

MODE 3

HYDROGEN COOLED OPEN CYCLE TURBINE SC GENERATOR

LIQUID HYDROGEN

HYDROGEN TO SPACE

ELECTRIC POWER 100 MW(e) MINUTES TO HOURS

FIGURE 2   Multimode reactor operation.

OUTER COATING
HIGH DENSITY PyC

KERNEL

INNER COATING
LOW DENSITY PyC

r_0
r_1 r_2

(a)

OUTER ISOTROPIC PyC

SILICON CARBIDE

KERNEL

INNER ISOTROPIC PyC

BUFFER PyC

r_1 r_0
r_2
r_3 r_4

(b)

FIGURE 3   BISO and TRISO fuel particles, developed by General Atomics.
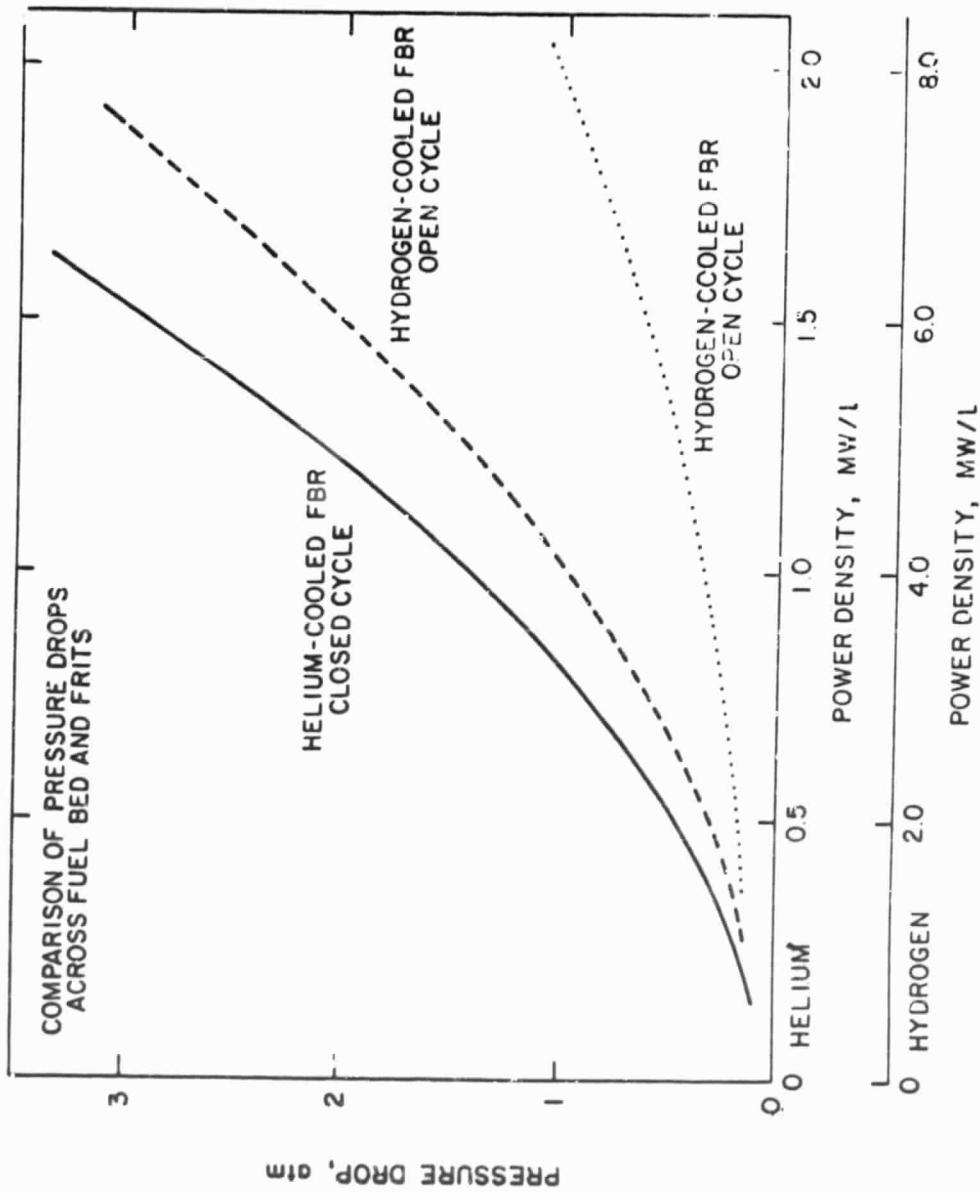
125
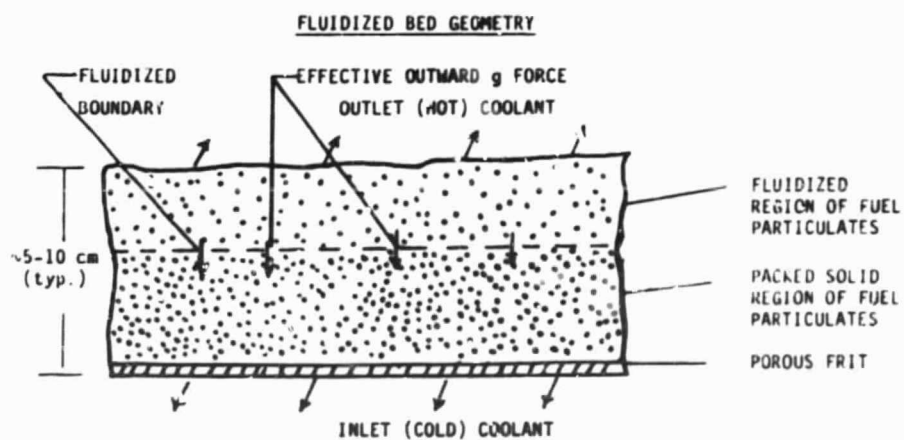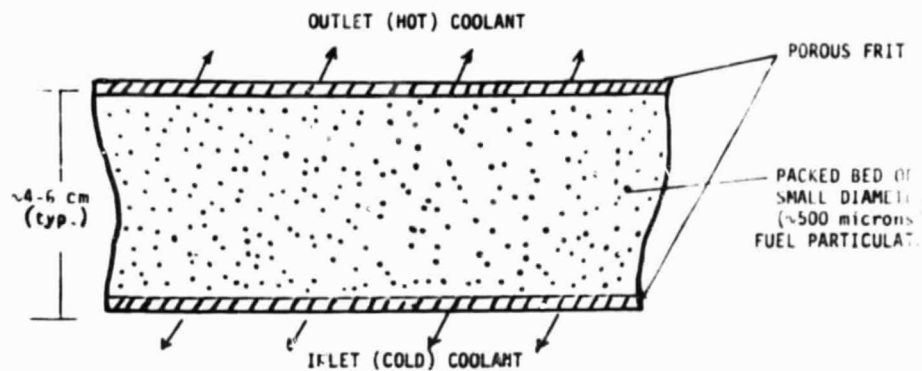
FIGURE 4   Fixed-bed geometry.

FIGURE 5 Power densities, in megawatts per liter.
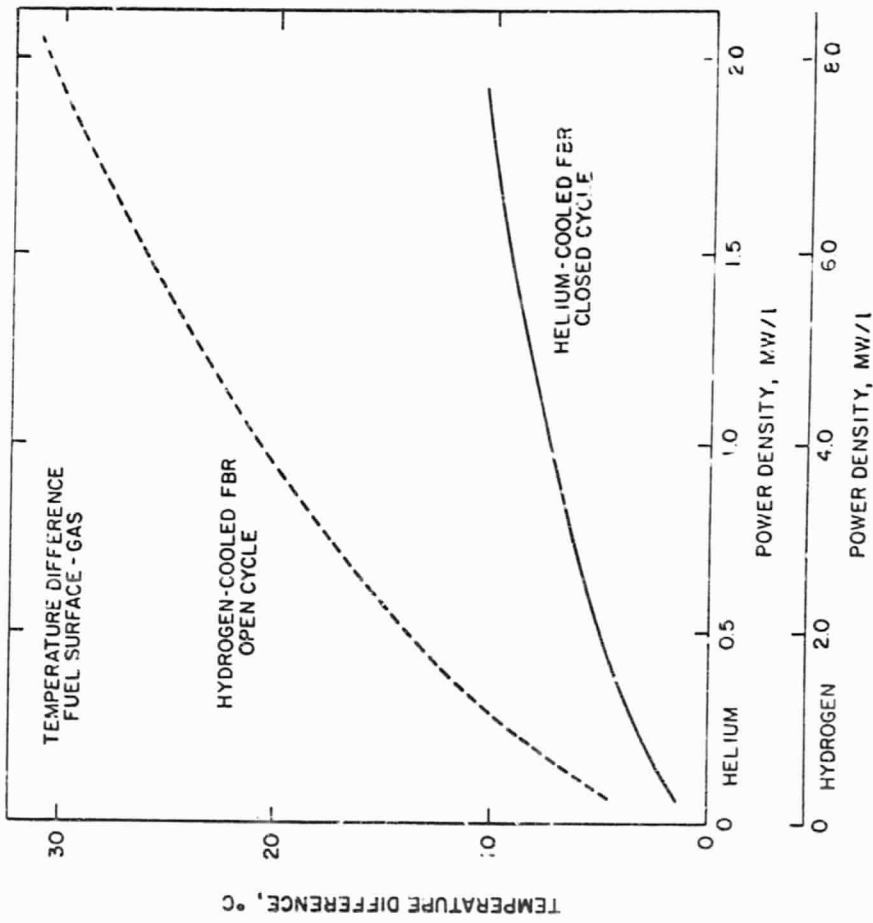
FIGURE 6 Average film ΔT between fuel particles and gas coolant as a function of power density.
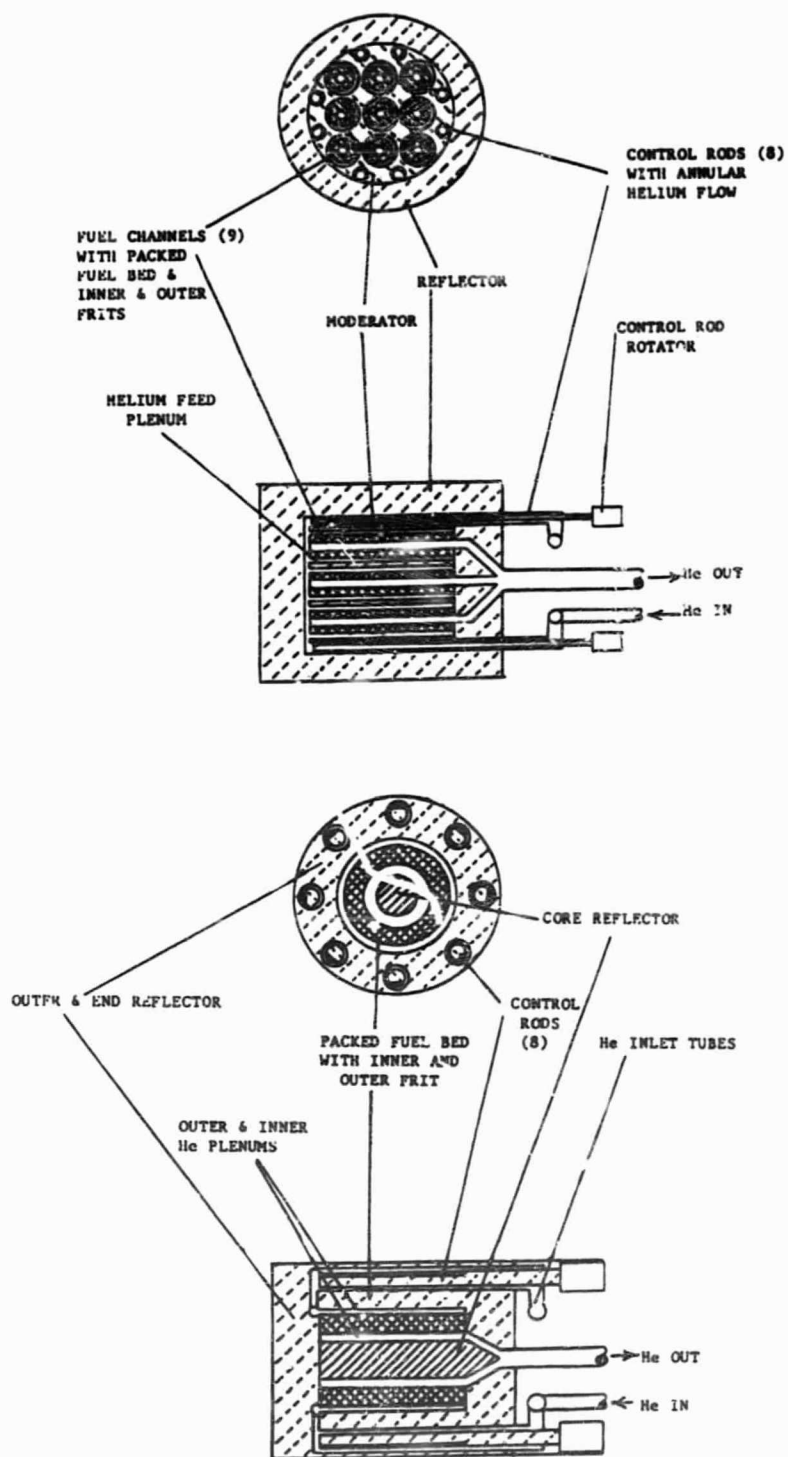
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FIGURE 7   Alternative configurations for the FBR.
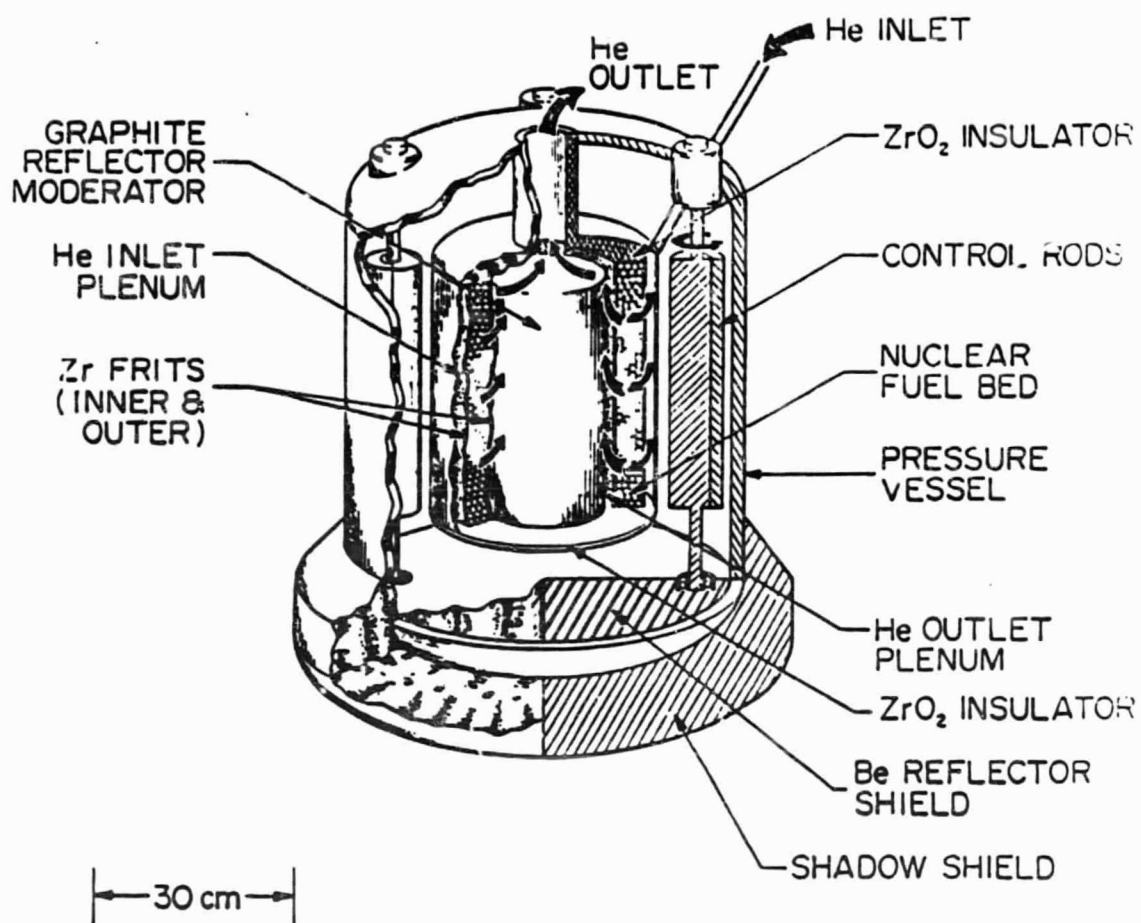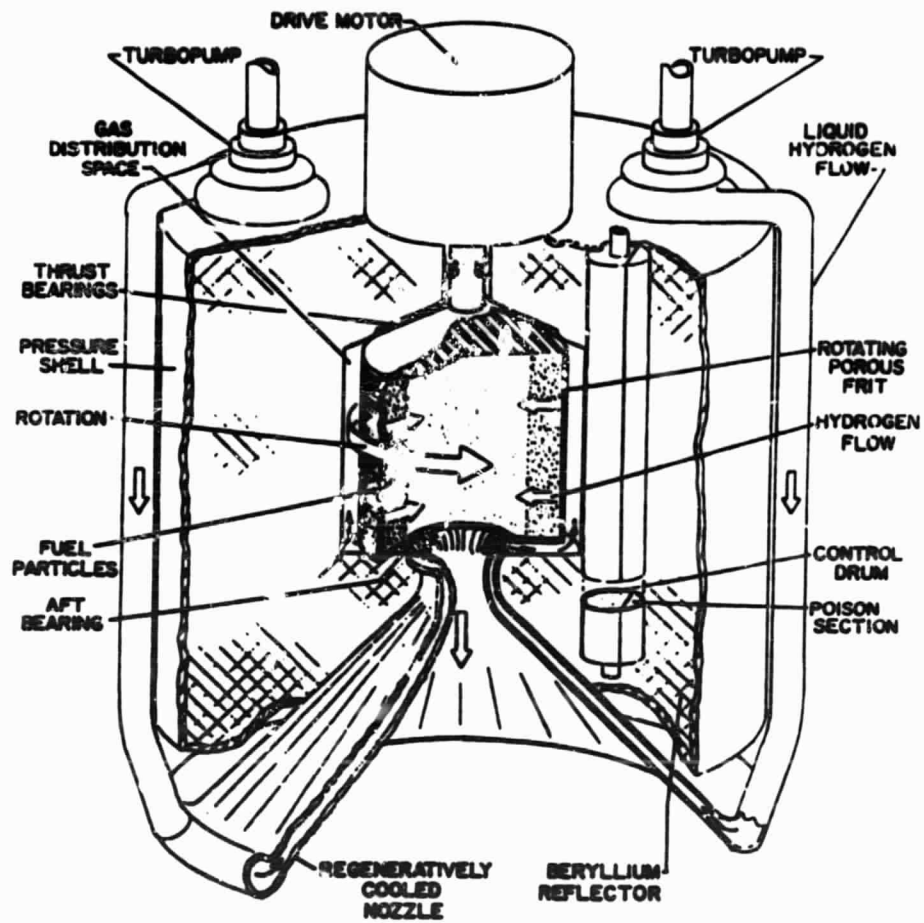
FIGURE 8   Fixed-Bed Reactor (FBR).

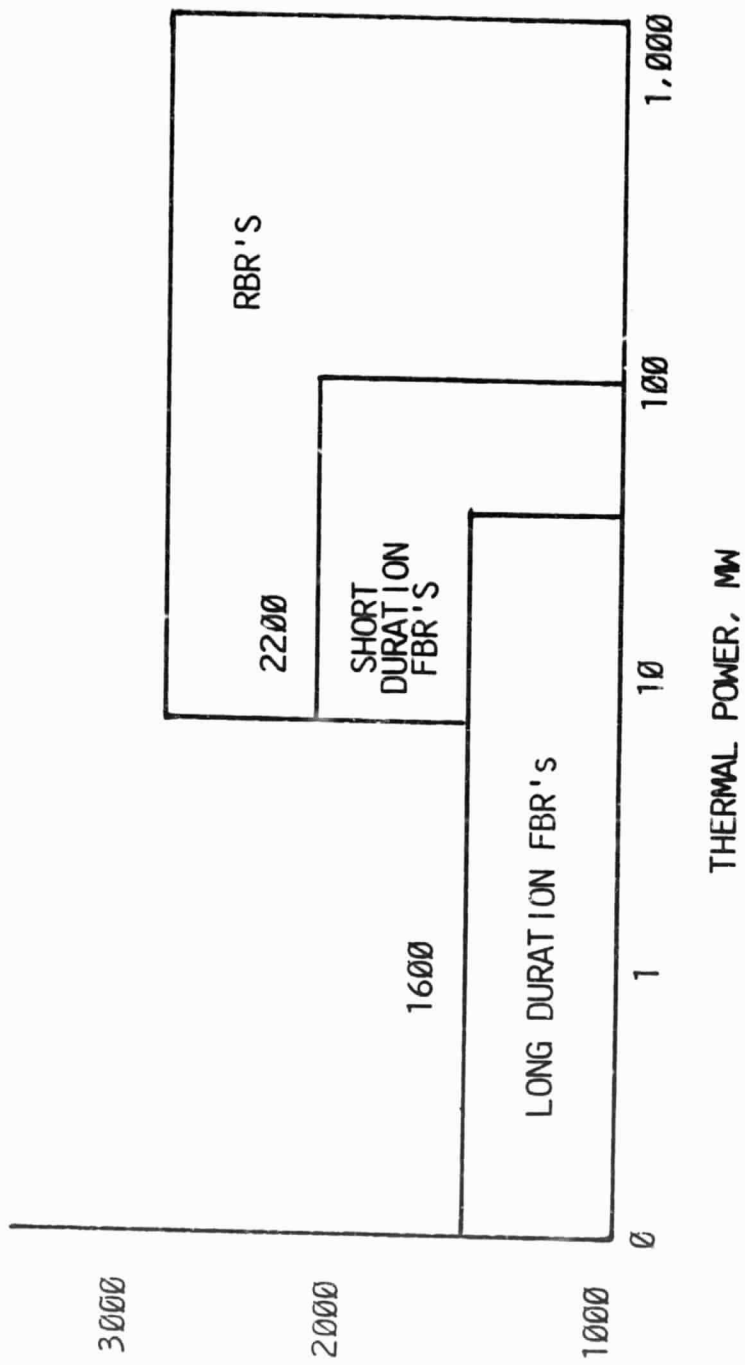FIGURE 9   Rotating fluidized-bed rocket engine.

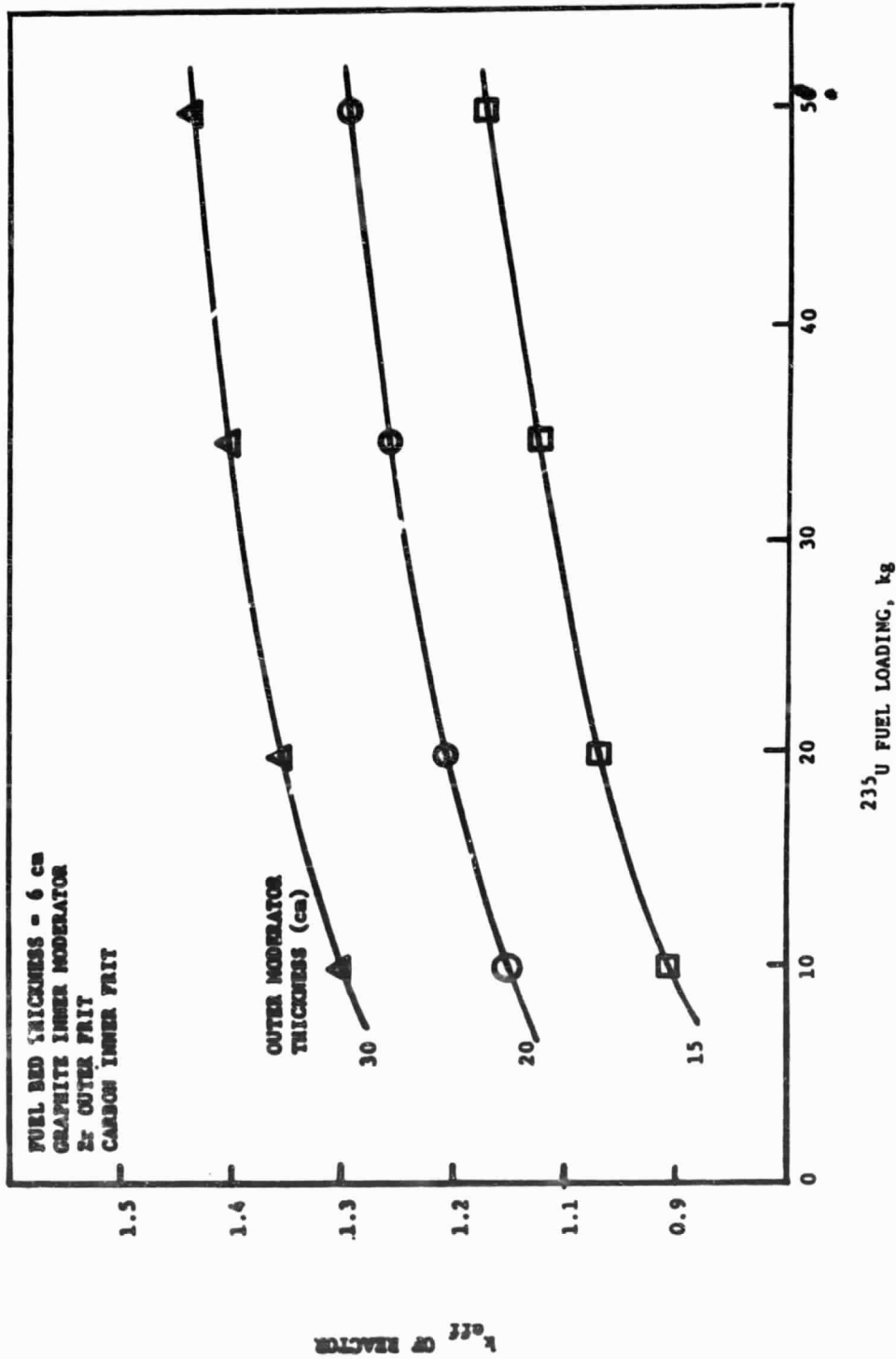FIGURE 10 Power regimes for FBRs and RBRs.

FIGURE 11   Fixed-Bed Reactors with Be moderator; fuel bed outside diameter is 50 cm.
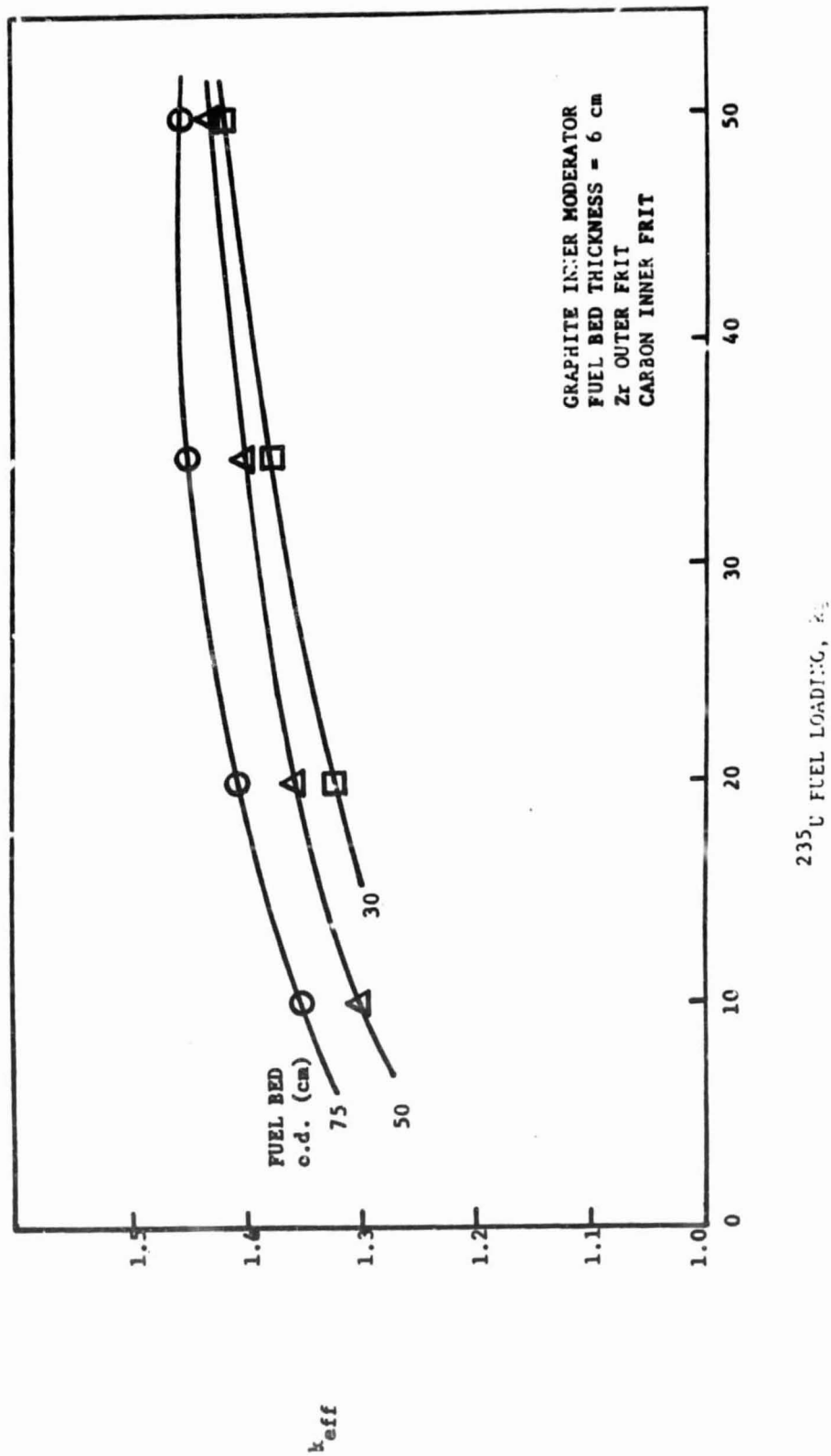
FIGURE 12 Fixed-Bed Reactors with Be moderator; outer reflector thickness is 30 cm.

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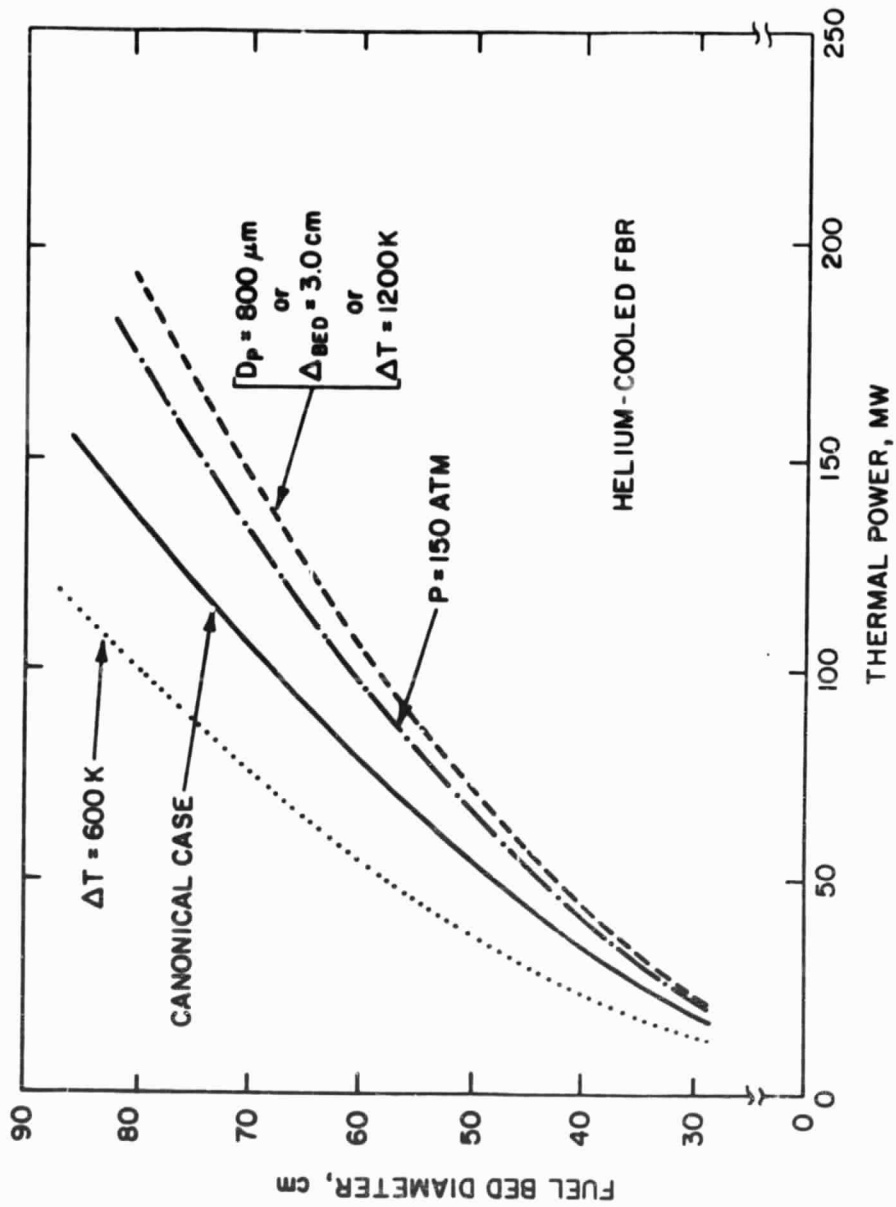
FIGURE 13 Helium-cooled FBR.
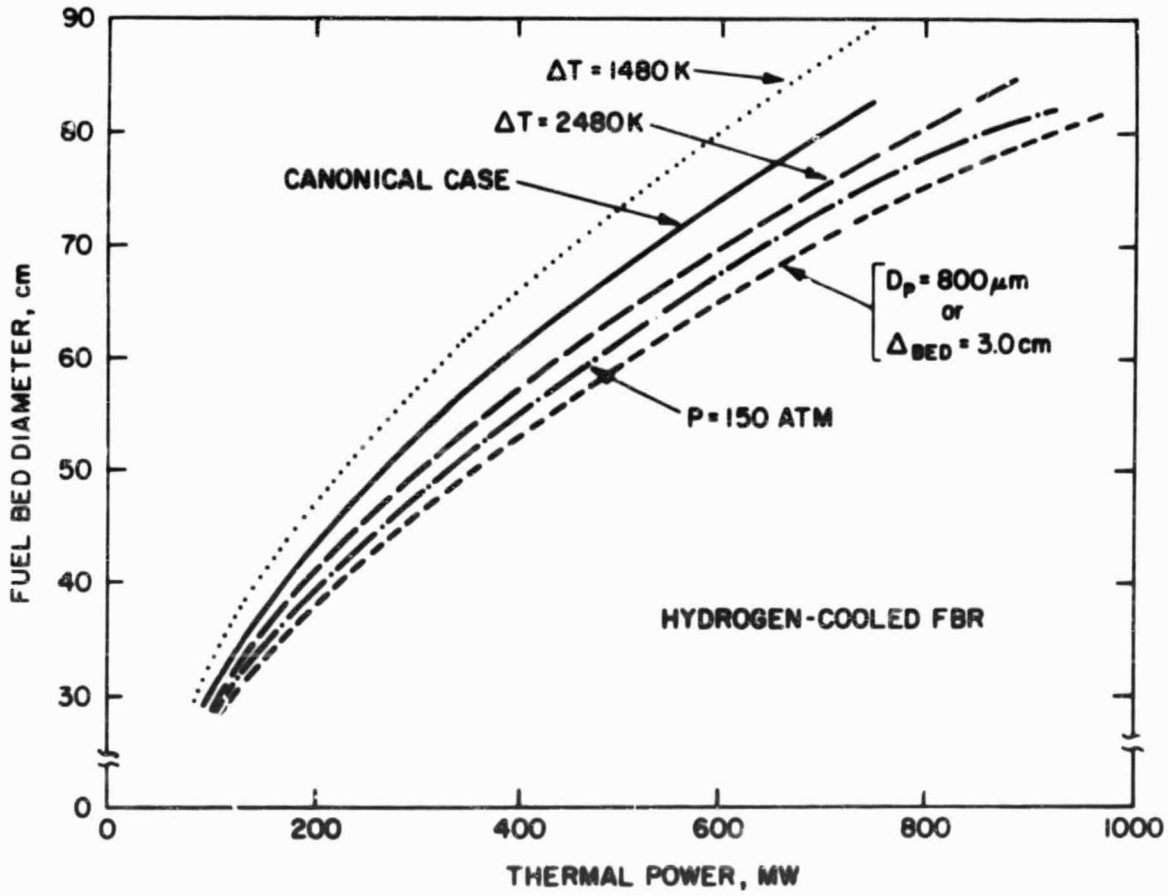
FIGURE 14 Hydrogen-cooled FBR.
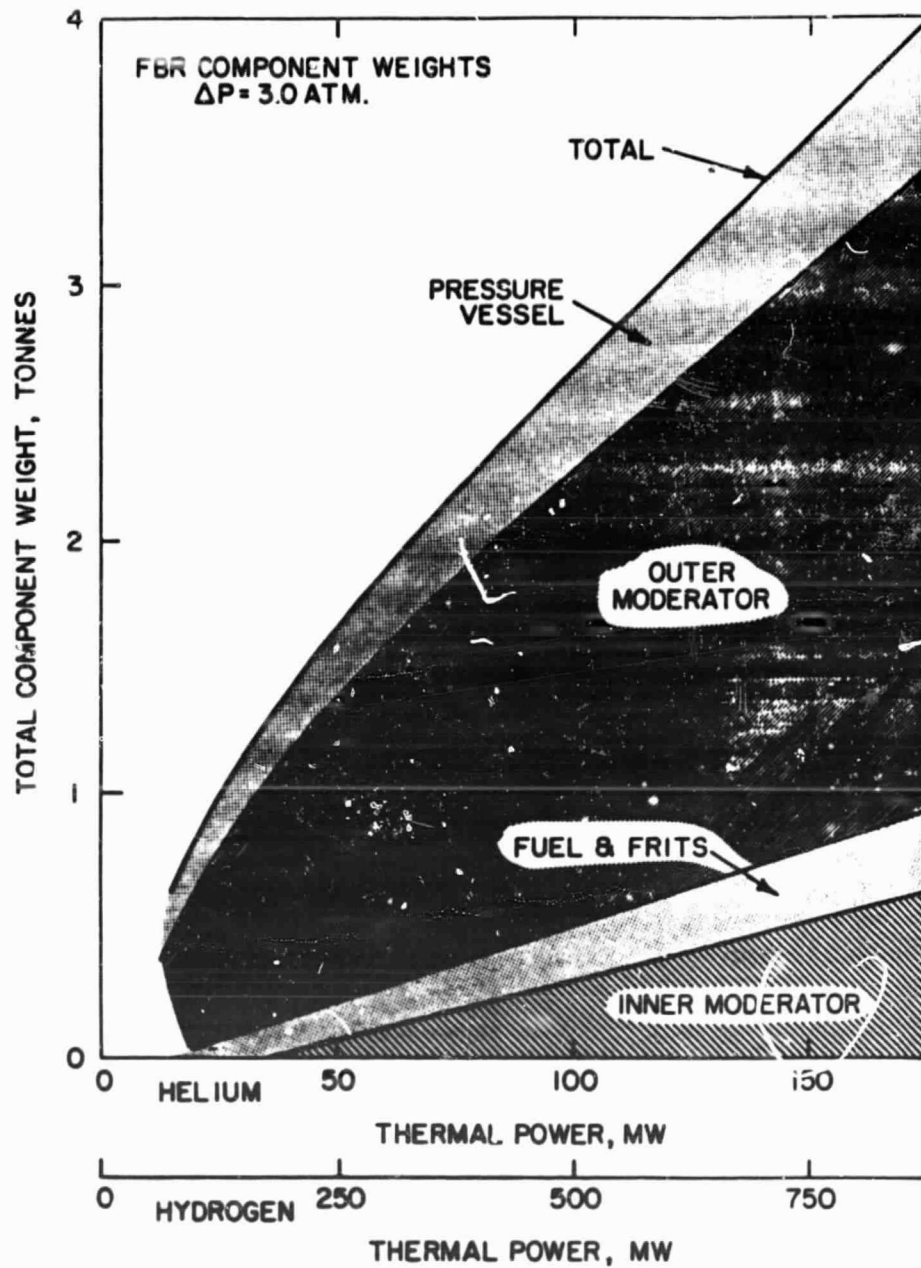
FIGURE 15 FBR component weights.
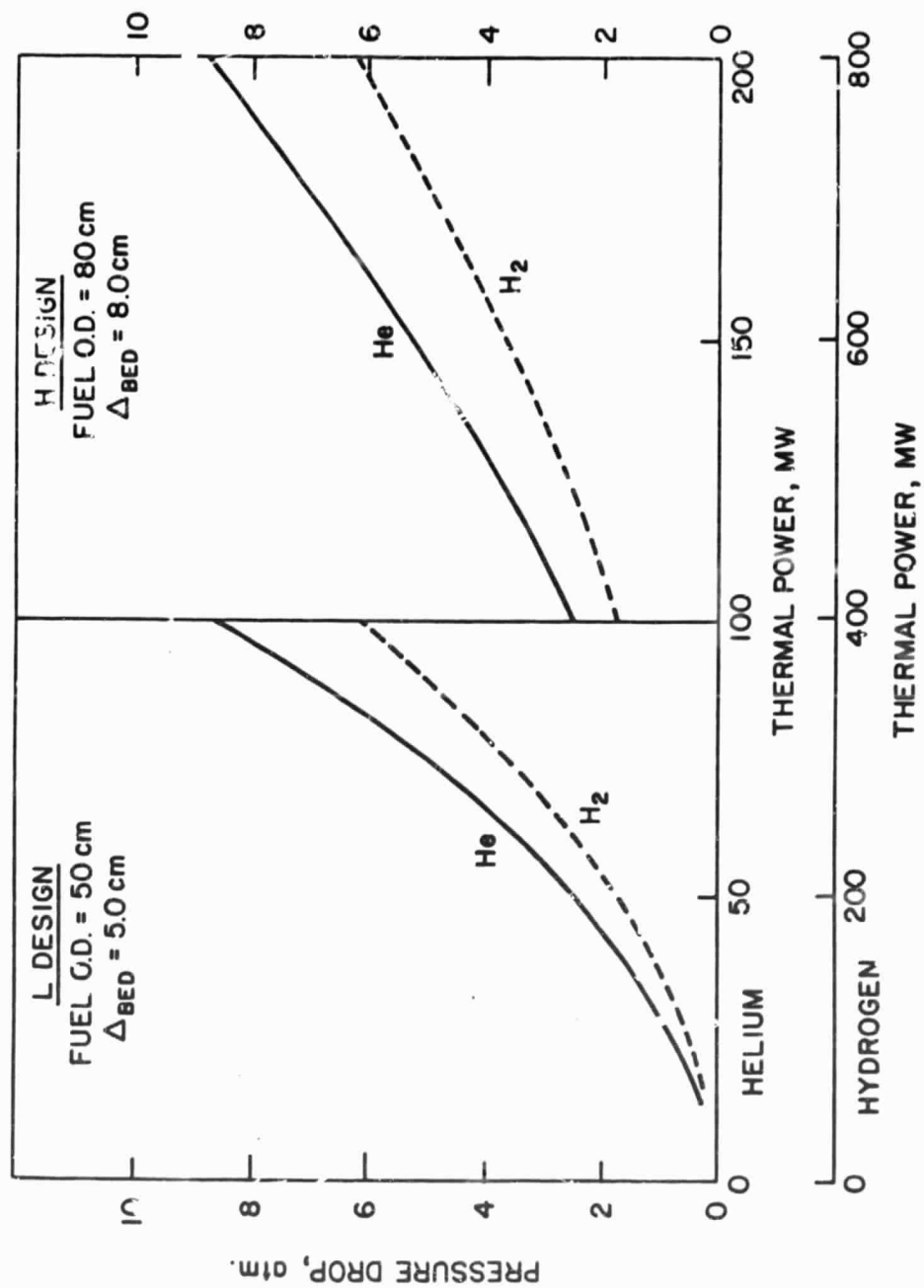
FIGURE 16 Pressure drop as a function of power for both He and H_2 coolant.
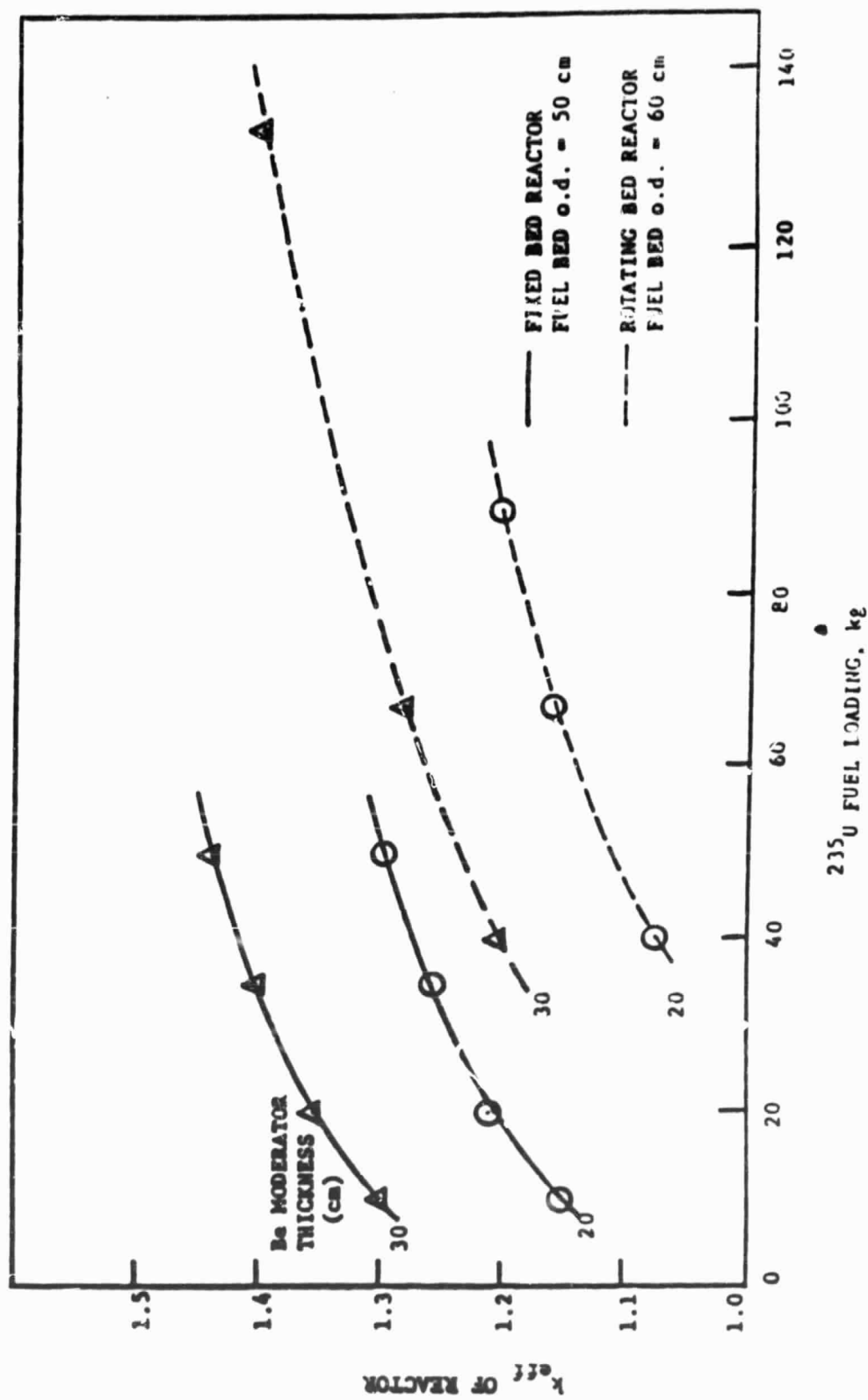
FIGURE 17 Comparison of critical masses for Fixed- and Rotating-Bed Reactors.
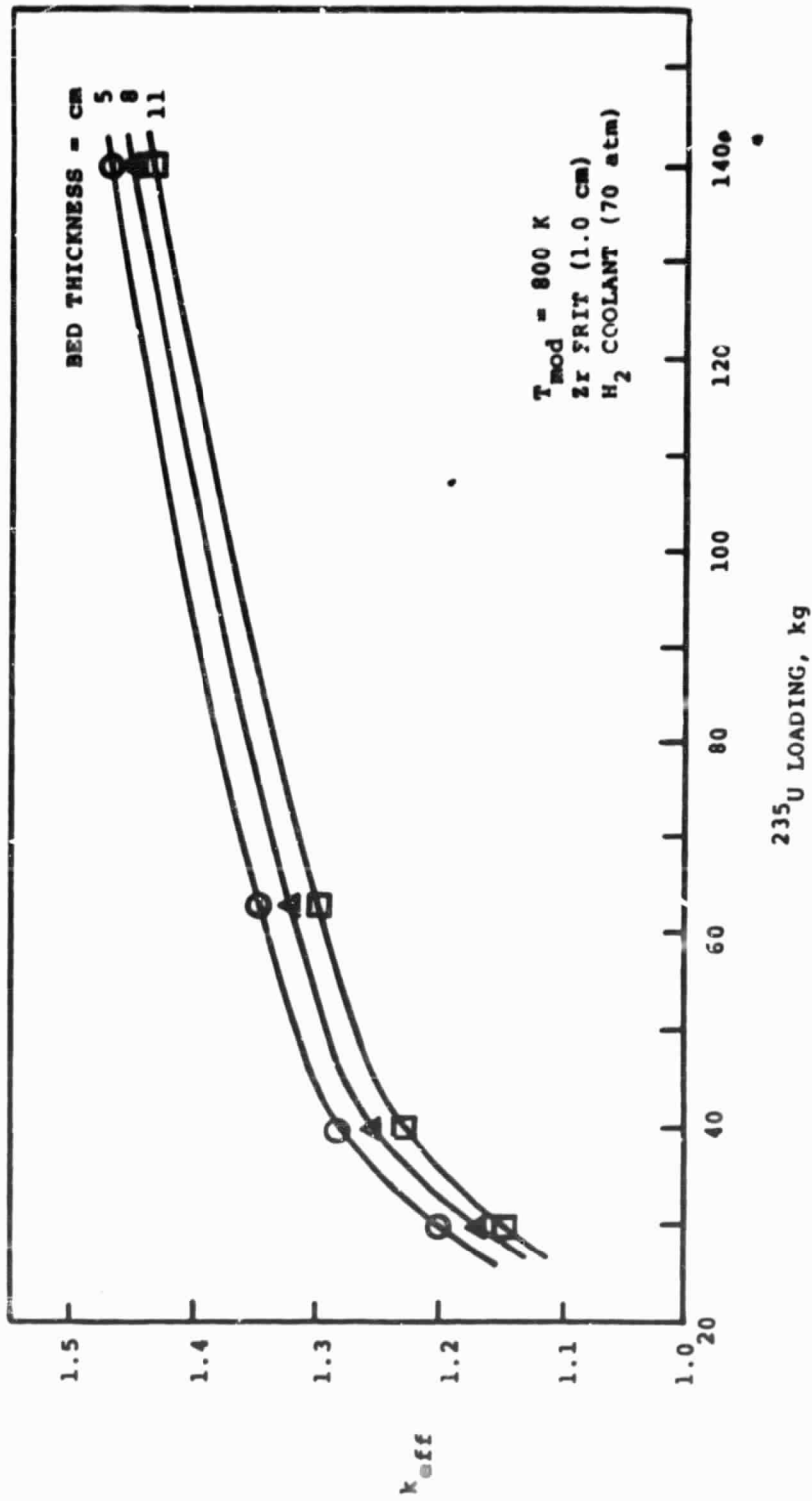
FIGURE 18 Rotating-Bed Reactor: k_{eff} versus uranium loading.
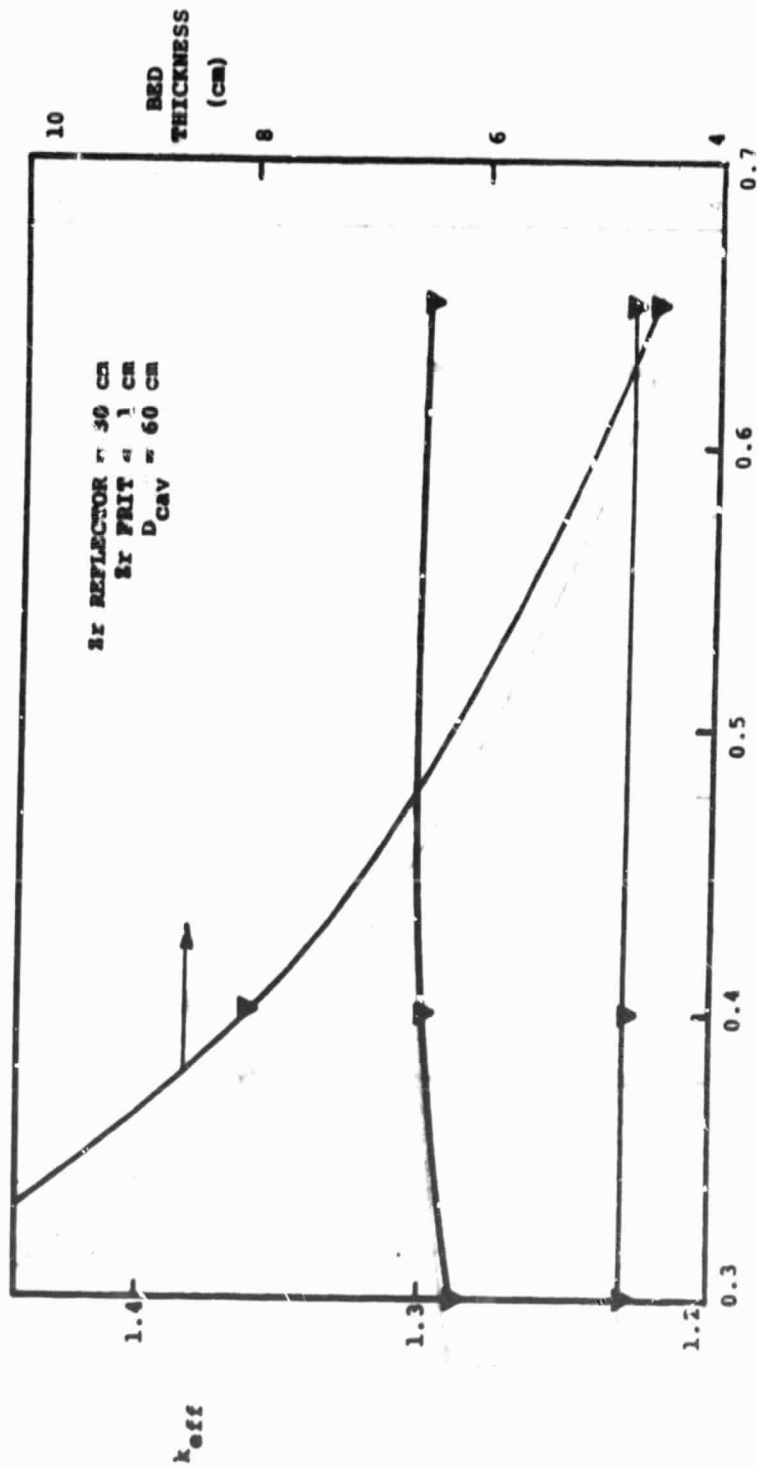
FIGURE 19 Criticality versus bed solid fraction.
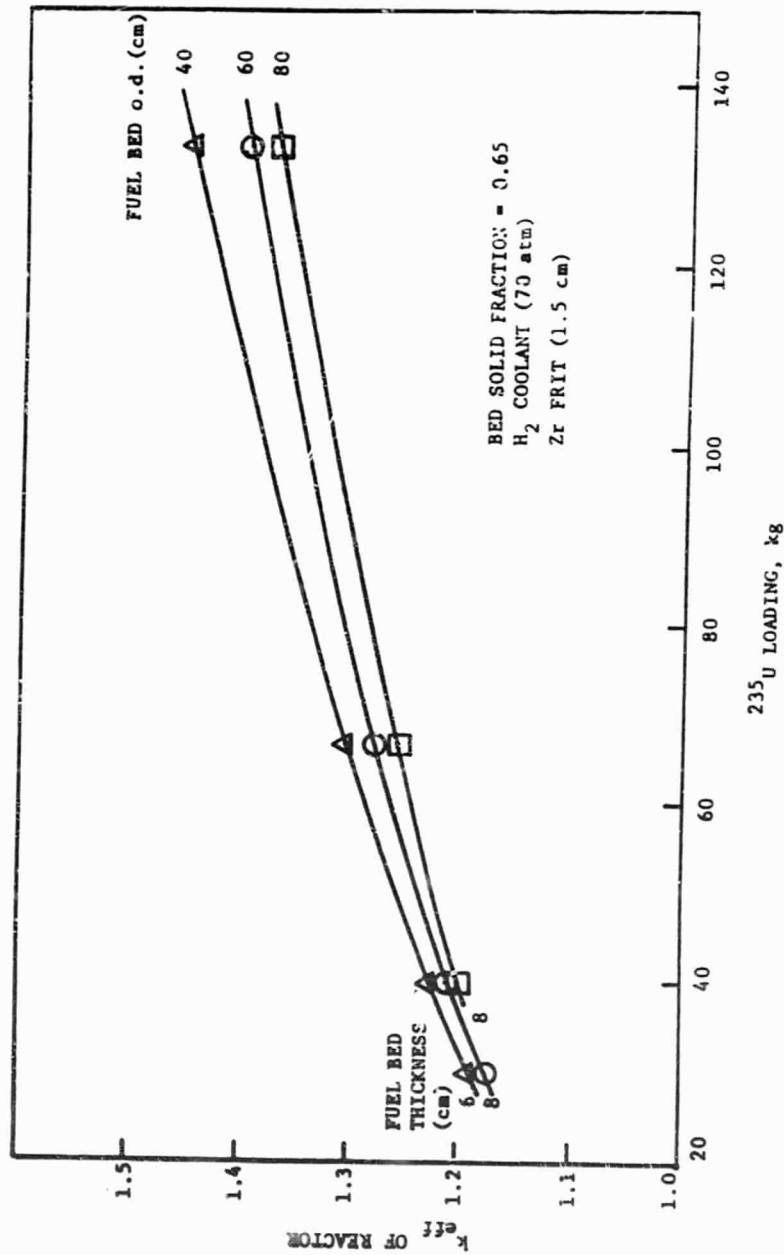
FIGURE 20   Rotating-bed reactor with Be moderator; moderator thickness is 30 cm, T_m is 80 K.
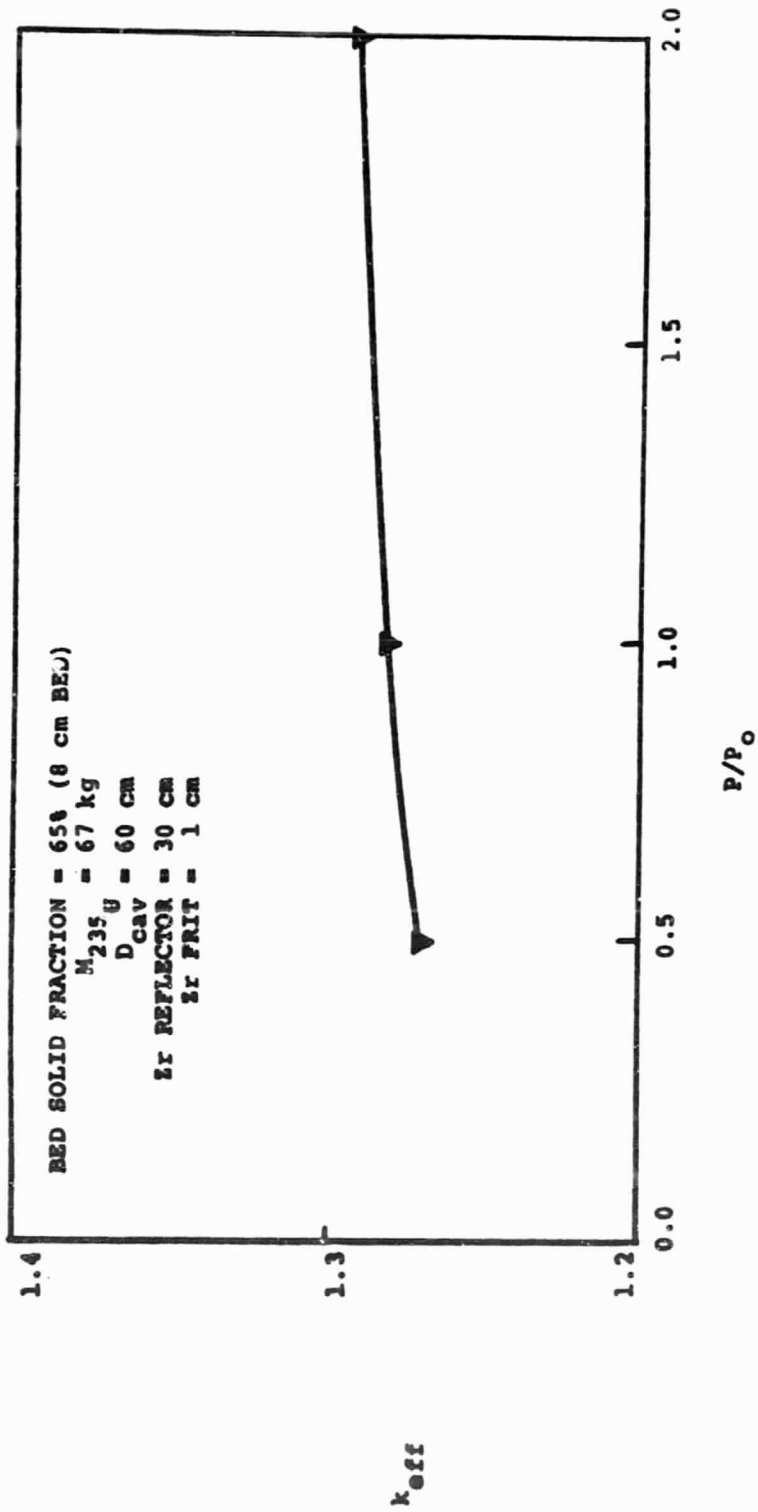
FIGURE 21 RBR k_{eff} versus H_2 cavity pressure.

EQUIVALENT DENSITY FACTOR OF Zr FRIT ($\rho=1$ CORRESPONDS TO NORMAL DENSITY).
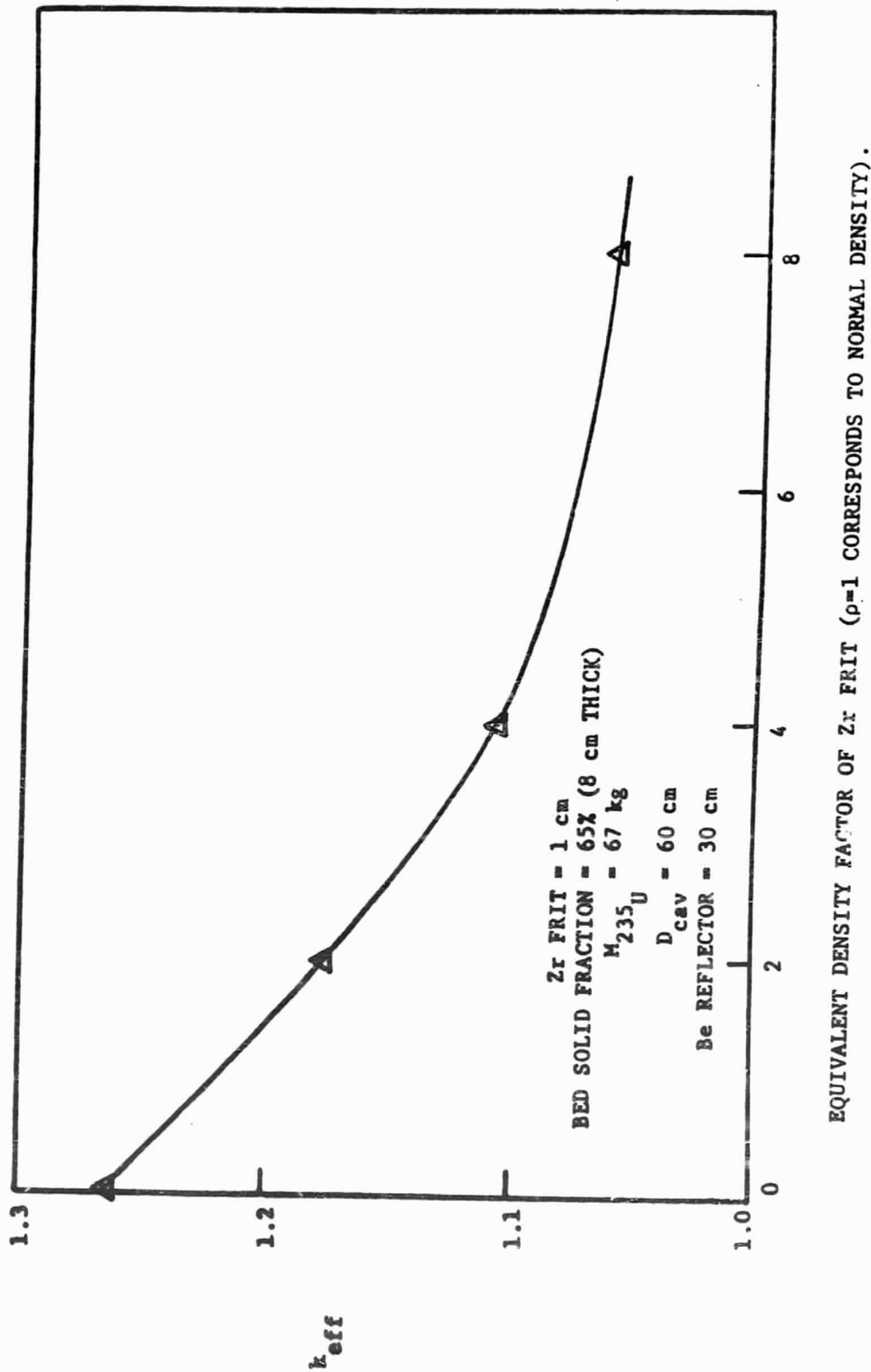
FIGURE 22 Effect of frit absorption on k_{eff} of RBR.

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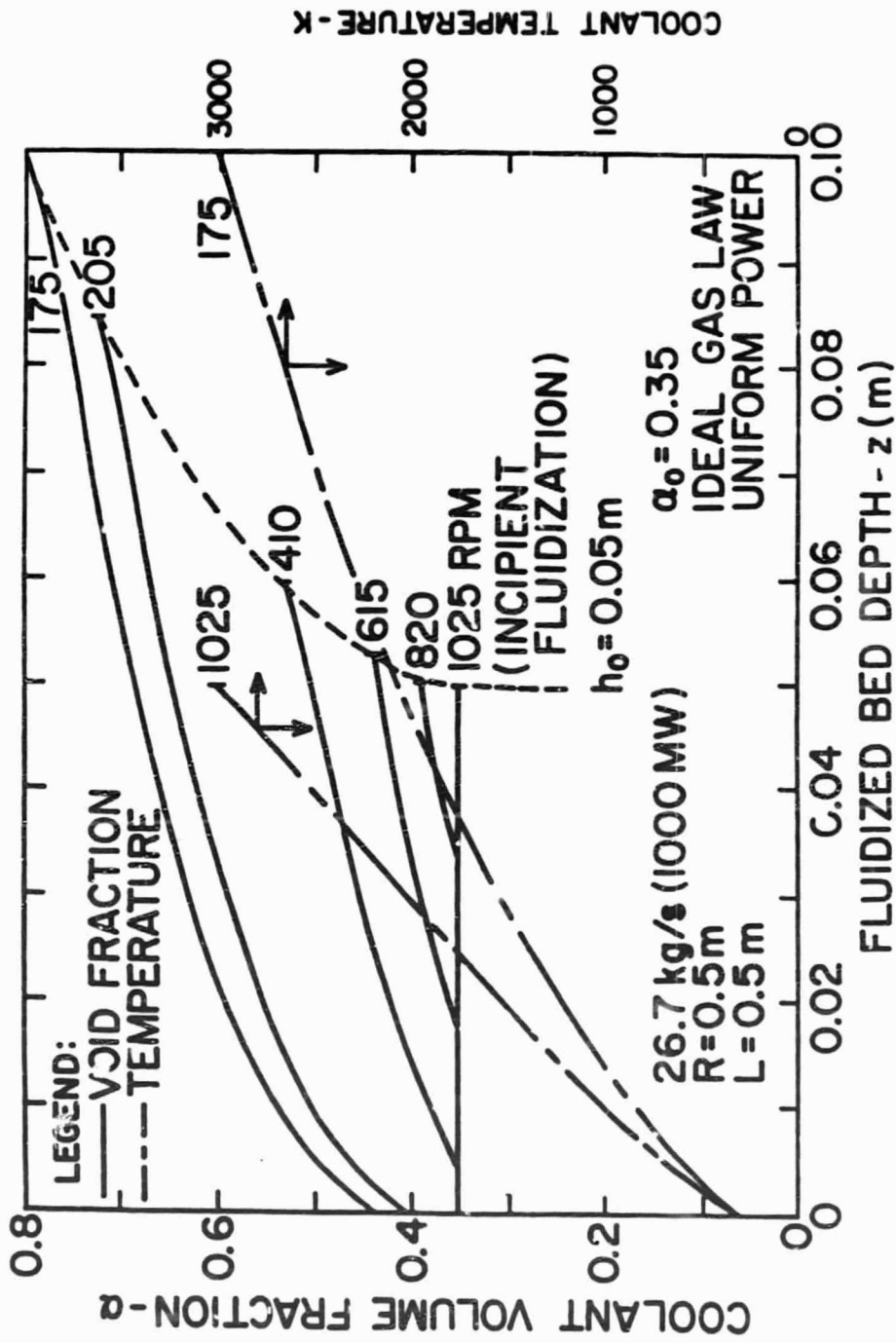
FIGURE 23 Coolant volume fraction and temperature as a function of fluidized-bed depth.

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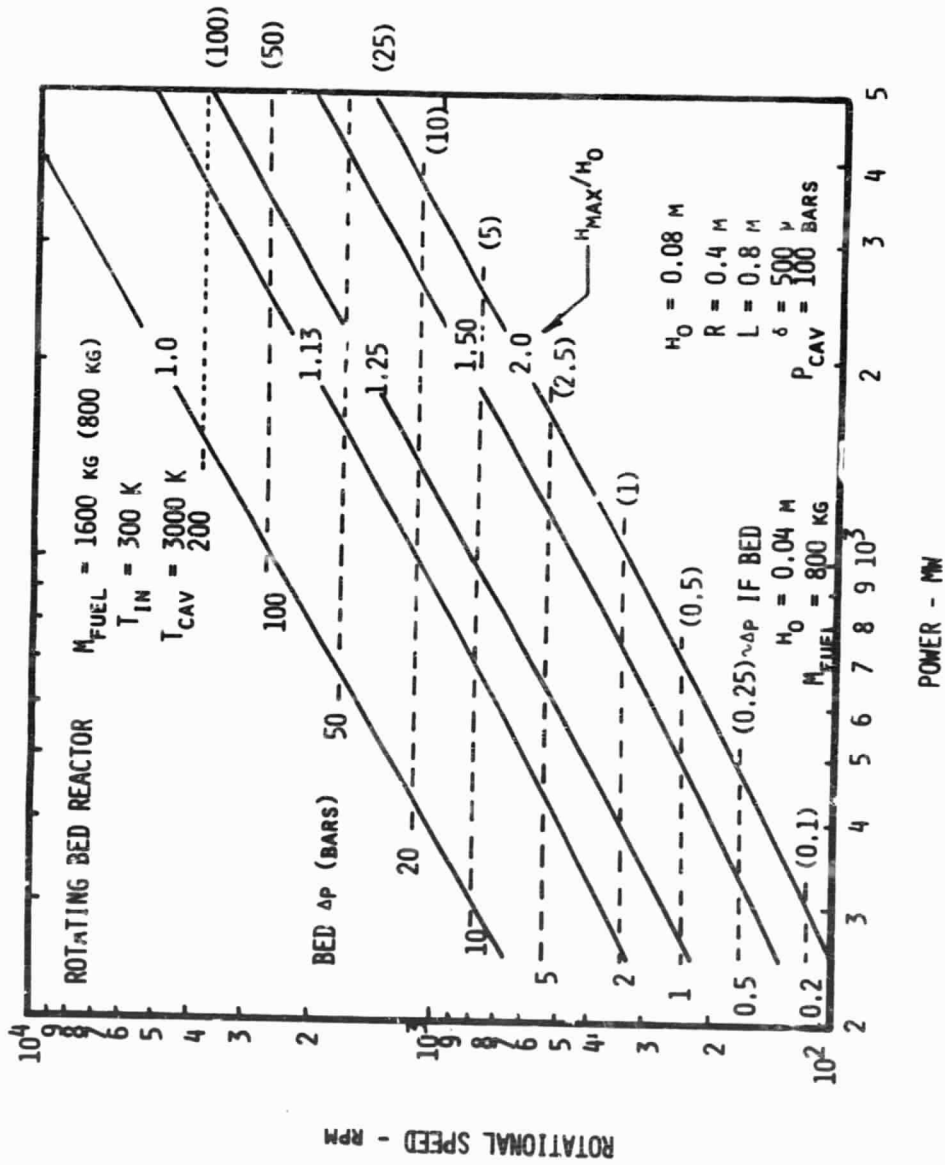
FIGURE 24 RBR rotation speed versus power level.
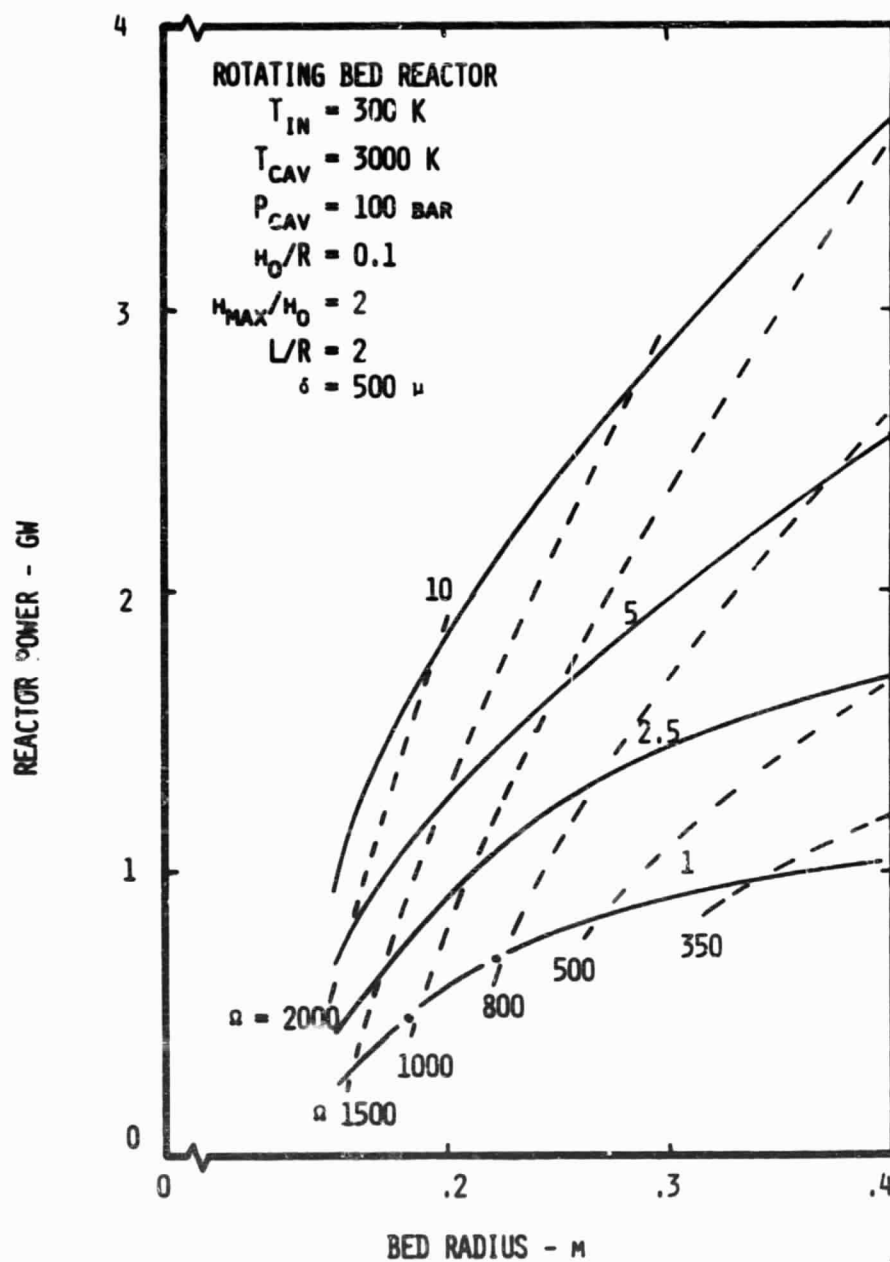
FIGURE 25 Reactor power as a function of radius for $H_{max}/H_O = 2$, and $H_O/R = 0.1$.

1 m.
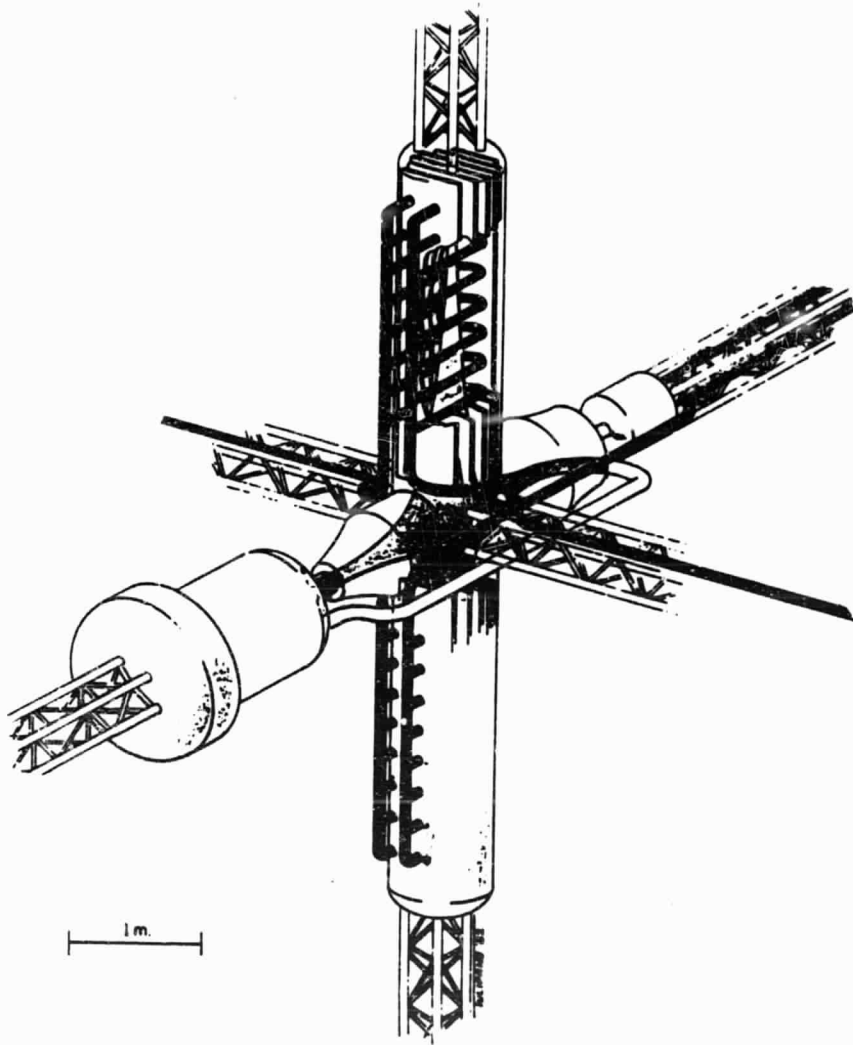
FIGURE 26   The 100-MW(e) closed-cycle system.
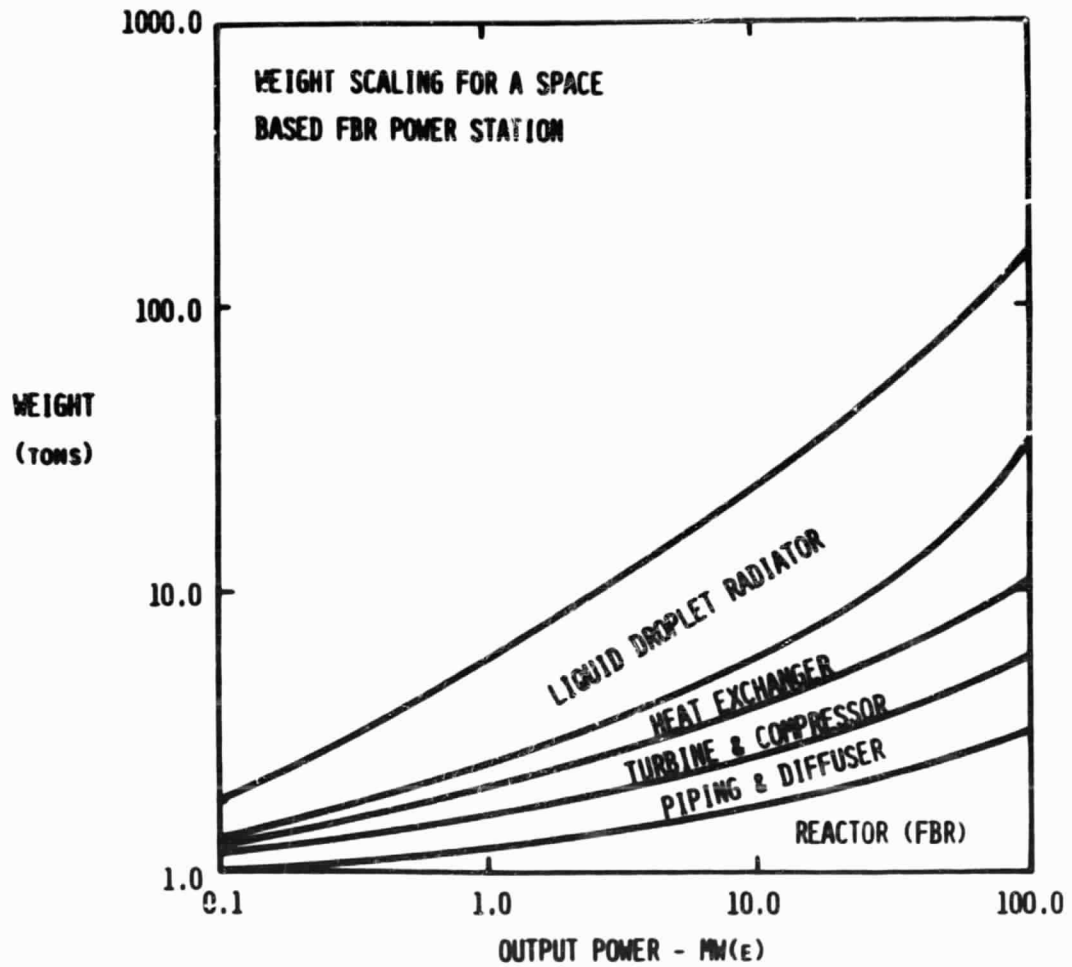
FIGURE 27 Weight scaling for a space-based FBR power station.

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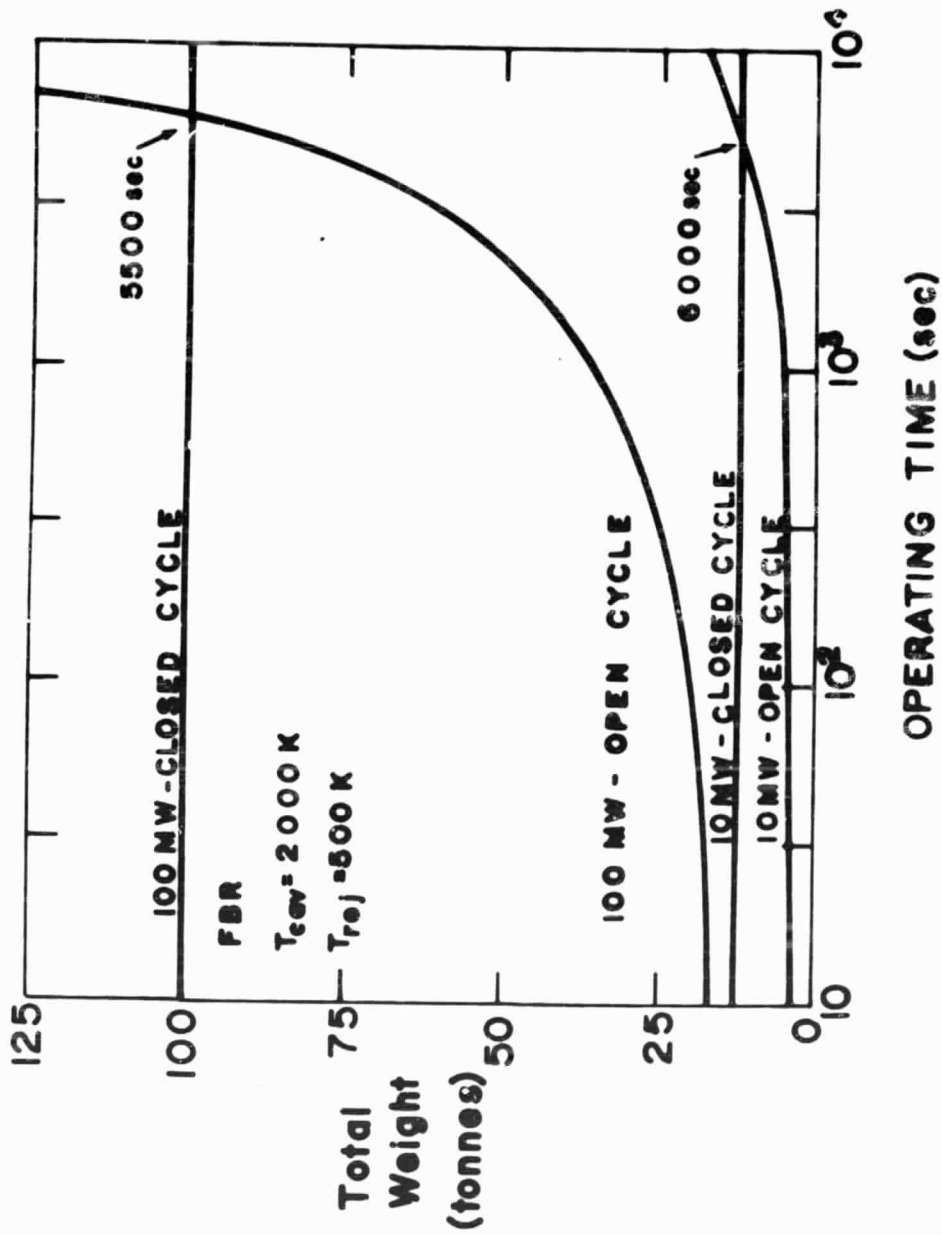
FIGURE 28   The 100-MW, closed-cycle operating time versus system weight (in tonnes).
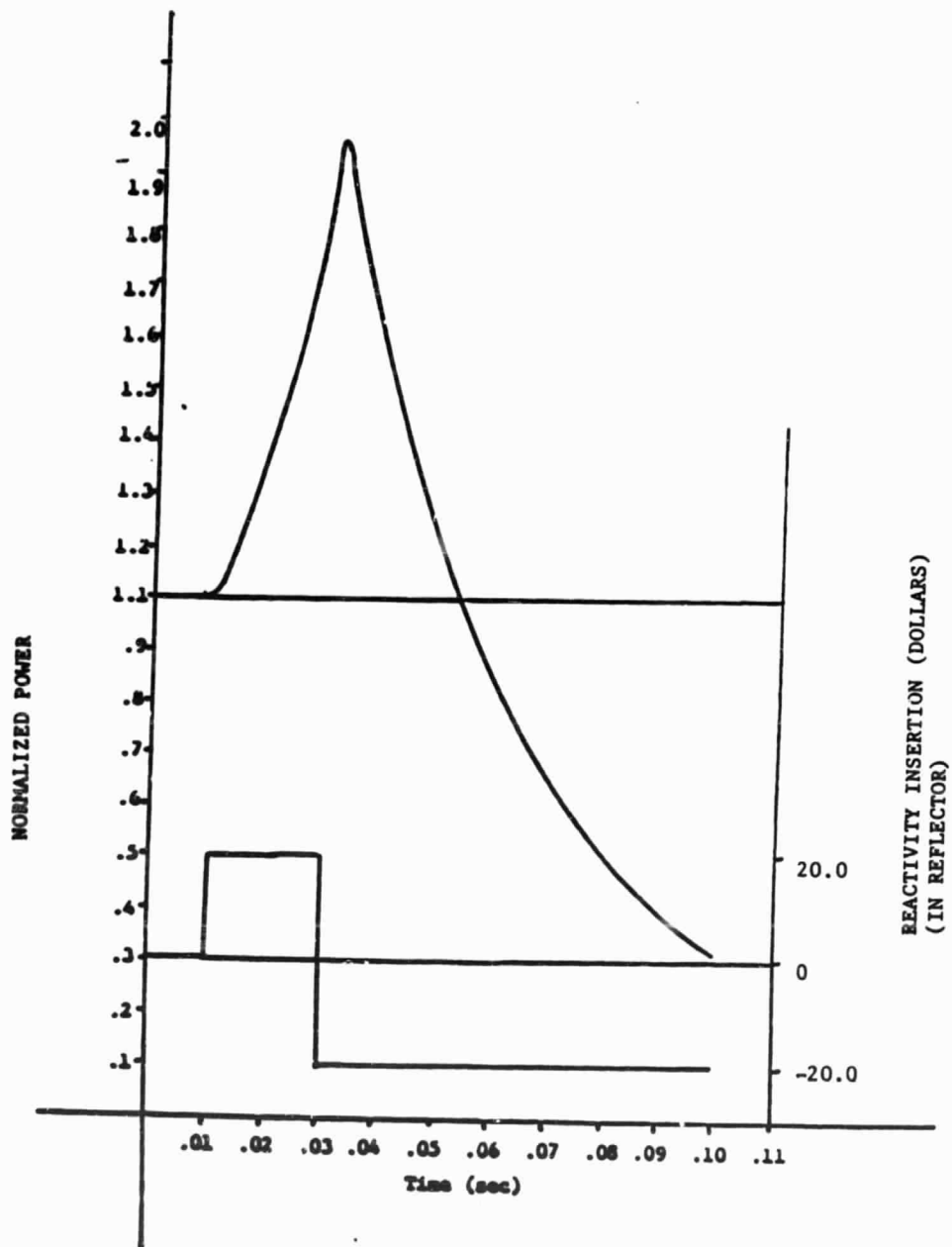
FIGURE 29 Variation of normalized power for a step insertion in the reflector of a RBR.

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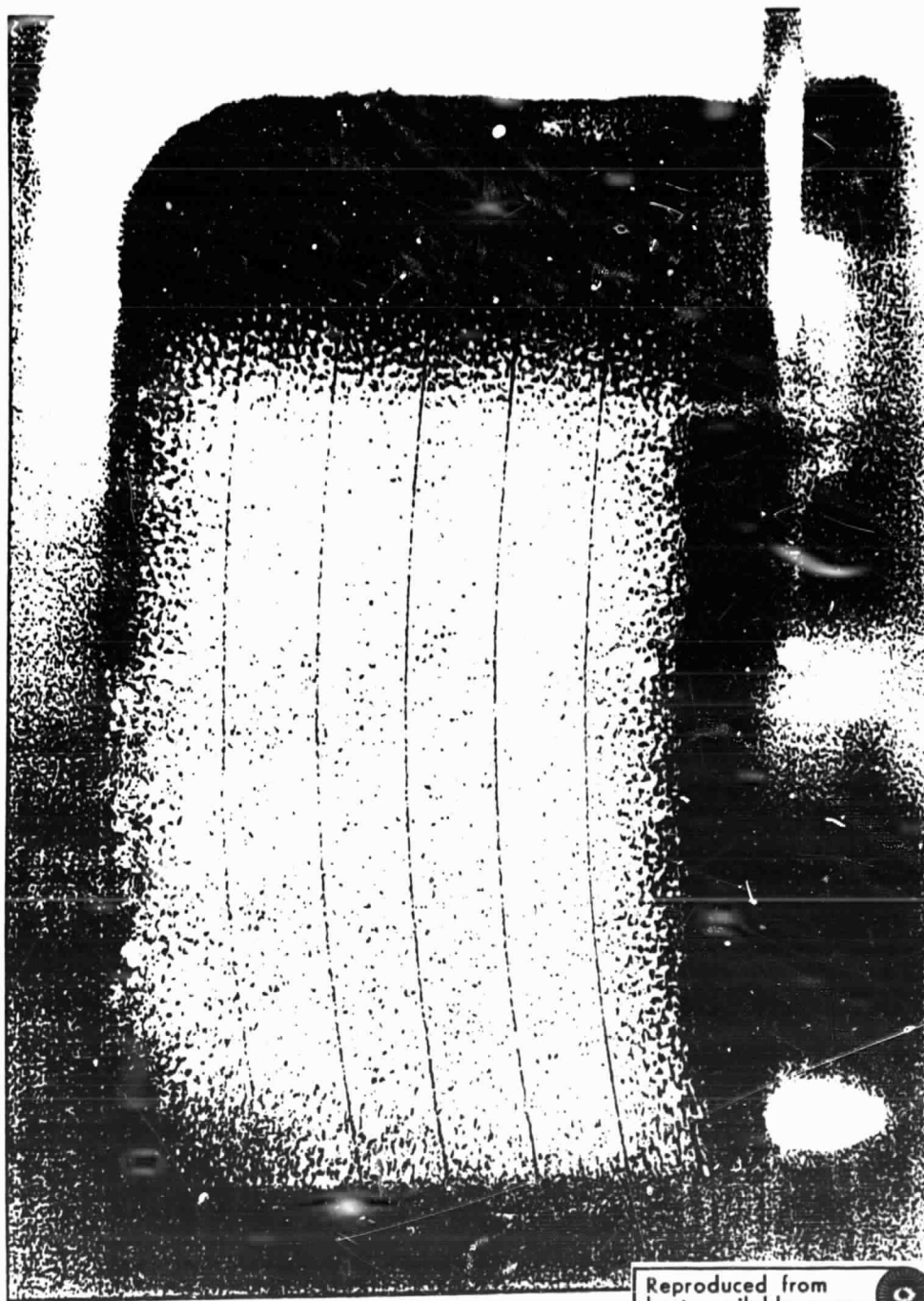
FIGURE 30 View of bed taken through bottom plate: 500-μm glass beads, 1,000 rpm, 170 m^3/min.

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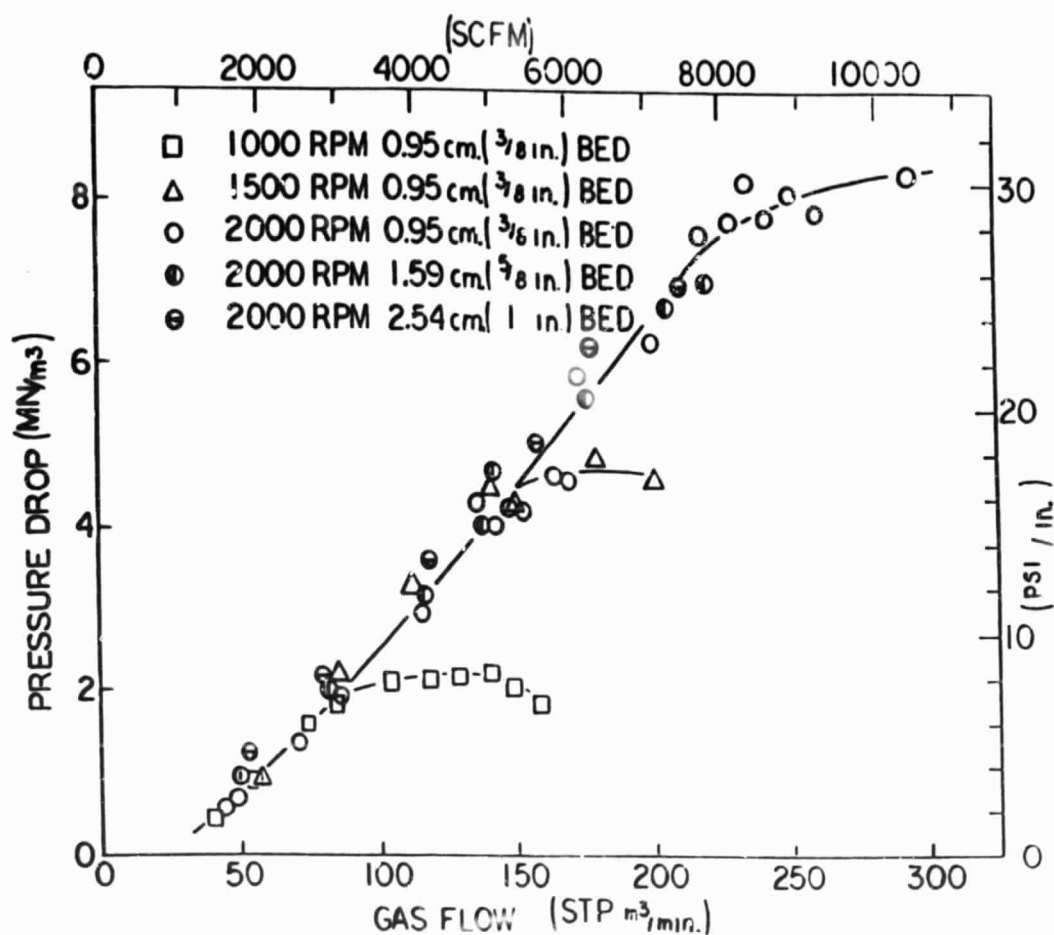
FIGURE 31 Correlation of fluidization data for 500-μm glass beads; specific gravity is 2.5.

FBR TEST ARRANGEMENT

STEEL VESSEL - 36" nom. dia. x 1" thk.

$S_1 = 15,000$ psi

$t = \dfrac{P_r}{S} = \dfrac{750 \times 18"}{15,000} = 0.90"$

GRAPHITE

CARTRIDGE HEATERS

440 V
3-PHASE

He

GRAPHITE

He

84 cm

40 cm

3 cm

2 cm

20 cm

40 cm

WIREWAY

PRESS.
TEMP.

FBR

He

CONCRETE STAND

He
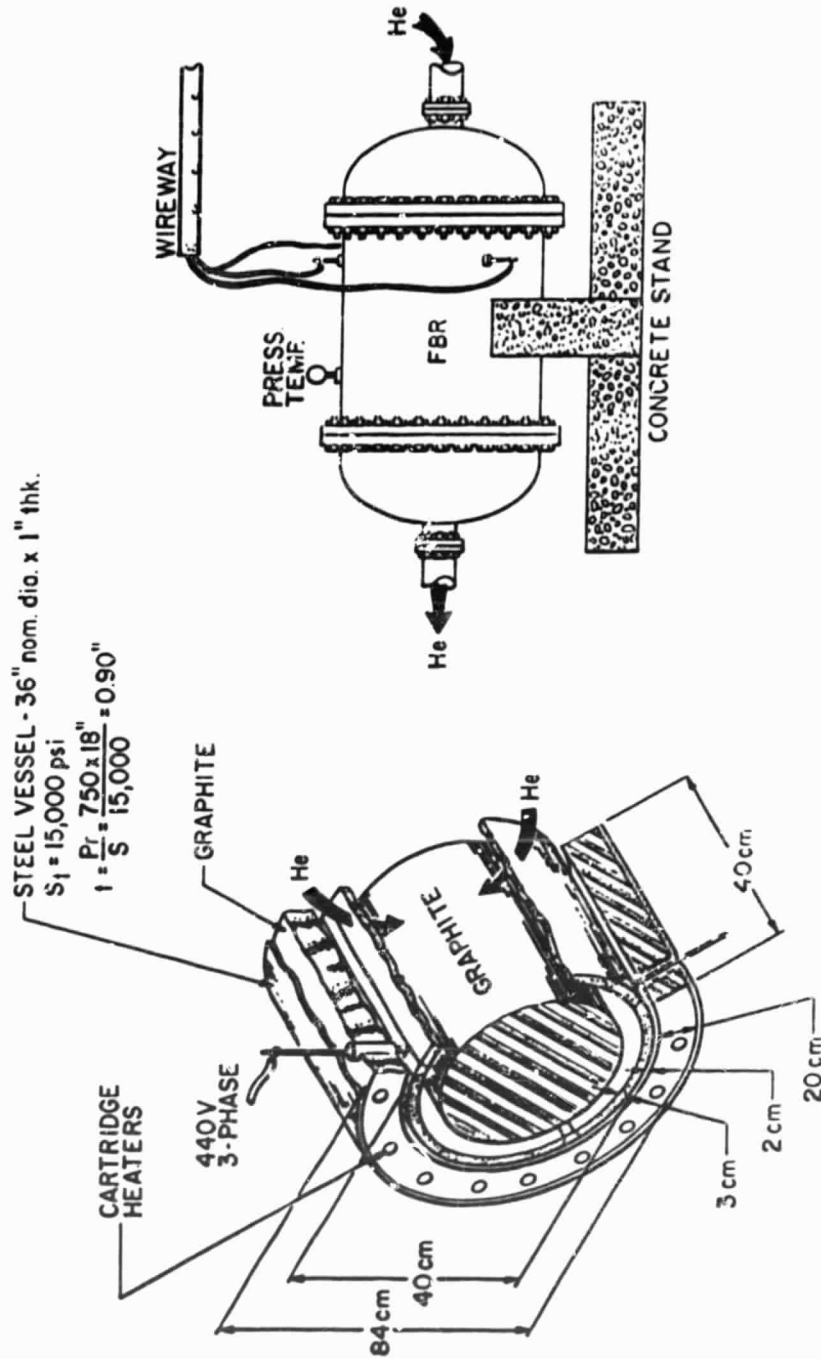
FIGURE 32   FBR test arrangement.

EXPERIENCE WITH GAS-COOLED AND LIQUID-METAL-COOLED HIGH-TEMPERATURE NUCLEAR REACTOR SYSTEMS

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Westinghouse Electric Corporation
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INTRODUCTION

The United States has made outstanding contributions to aerospace technology over the past 25 years. which have permitted dramatic progress in the exploration and exploitation of aerospace for the benefit of the civilian and military sectors. The successful commissioning of the Space Shuttle and the consequential interest in deploying larger and more advanced aerospace systems have called attention to the need for timely development of advanced power supplies so that important aerospace missions of the 1990s will not be unduly constrained by power considerations. Clearly, the most cost-effective and productive step in this direction is to assess the current status of applicable technology, followed by a developmental program that maximizes technology transfer and produces a reliable power system for a broad range of aerospace missions. Toward this objective, this paper provides an overview of relevant aerospace nuclear power and propulsion programs in which Westinghouse played a significant role.

LIGHTWEIGHT HIGH-TEMPERATURE REACTOR EXPERIENCE

Propulsion Systems

Nuclear Rocket Engine (NERVA) (Westinghouse Astronuclear Laboratory, 1972)

Major progress in the development of lightweight, gas-cooled reactor technology was accomplished under the U.S. nuclear rocket program, which was initiated by the Los Alamos Scientific Laboratory (LASL) in 1955 and reached advanced development status through the efforts of a government-industry team. This program sought to utilize the high specific impulse of a nuclear rocket to circumvent the limitations of chemical systems for advanced scientific and civilian missions. The NERVA flight engine reactor was sized to provide 1,500-MW thermal

power for a rated engine thrust of 75,000 lb at a specific impulse of
825 s. The overall engine reliability requirement was confidently set
at an extremely high level: 0.995 at a 90 percent confidence level.

Based on the LASL KIWI design, the NERVA engine utilized a
hydrogen-cooled, epithermal reactor containing a core of uranium
carbide fuel beads dispersed in a graphite matrix and a radial
beryllium reflector with rotating neutron-absorber drums for control
of reactor power, as shown in Figure 1. The nucleonics of the reactor
was dominated by the large void fraction associated with the numerous
coolant channels and by the niobium coatings in the channels that
protected the graphite core against the highly reactive hydrogen
coolant.

In addition to demonstrating the ability of the reactor to operate
for an extended period at desired power and temperature levels, the
NERVA program had to address a number of basic feasibility questions
with regard to reactor structural integrity, restart capability, and
controllability. Structural integrity and performance were
demonstrated by successful operation of 15 KIWI and NRX reactors.
Early in the nuclear rocket program, serious vibrational problems were
encountered in the 1962 KIWI-B test, leaving the structural integrity
of the reactors in doubt. Tests conducted in 1964 by LASL and the
NERVA team proved that the difficulty was understood and had been
corrected by an improved lateral support system. Confidence then
increased sufficiently to increase the reactor performance goal to 60
min of full-power operation, which was subsequently achieved with the
NRX-A6 reactor test article.

In the NRX test reactors, H_2 entered the reflector from the
regeneratively cooled thrust nozzle at a temperature of the order of
-260^OF. After cooling the reflector, pressure vessel, control drum
actuators, internal shield, and support plate, the hydrogen entered
the core at approximately -230^OF. Most of the hydrogen passed
through the fuel element channels, wherein it was heated to
approximately 4000^OF. A fraction of the core hydrogen flow was used
to cool the tie-rods, after which procedure it was mixed with the fuel
channel exit gas to provide nominal nozzle chamber conditions of 550
psia and 3630^OF.

More than 25 reactor start-ups, including many restarts, were
successfully accomplished in the KIWI/NRX test series. In addition,
10 reactor start-ups to power conditions were successfully executed in
the first complete engine system test (NRX/EST). These tests
conclusively demonstrated the restartability and reusability of the
NERVA nuclear system.

Other reactor tests resolved the questions about controllability of
the system associated with the nuclear reactivity effects of the
hydrogen coolant under the extremes of temperature, flow, and start
times dictated by a nuclear rocket.

Tests then shifted from reactor-related experiments to the
performance of a ground test engine called the NERVA-XE, successfully
completed in September 1969. The tests were conducted with the engine

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firing downward into a simulated space environment, as shown in Figure 2. The steps included multiple restarts, throttling, and automatic control start-up by use of "bootstrap" techniques. The engine was started 28 times and was operated at various thrust levels for a cumulative time of 3 hours and 48 min, which included some time at full power of 1,100 MW and thrust of 55,000 lb.

The technology development phase of the NERVA program was successfully completed with the NRX-A6 reactor test and the XE-Prime engine test. The feasibility questions had all been answered satisfactorily, the necessary development information had been obtained, and reactor endurance capability of greater than 60 min at full temperature and power had been demonstrated. Development of a flight engine was ready to proceed in 1971, when the NERVA program was terminated.

Nuclear Aircraft

Nuclear Extended Range Aircraft (NuERA) (Muehlbauer and Thompson, 1972). This study of the U.S. Air Force was initiated in 1966 and had the basic objective of assessing the feasibility of manned nuclear aircraft propulsion using state-of-the-art reactor technology. The reference design parameters that guided the study included a reactor plant power level of 275 MW, a reactor outlet temperature of 1800°F, a design lifetime of all reactor components (including core) of 10,000 hours, and a turbine inlet air temperature of 1600°F.

Concurrent with the design of the reactor system was the evaluation of system safety during normal and abnormal aircraft operations, including the potential consequences of a crash. In pursuing the high-temperature, liquid-metal-cooled reactor technology specified for this study, various design options were considered. A single-phase heat transport system was chosen owing to several disadvantages of a two-phase system, particularly the larger flow-area requirements that would increase component sizes and weights. Although direct transport of the reactor coolant to the engine turbines promised higher thermal efficiency, a primary/secondary heat transport system was elected to satisfy safety requirements and to avoid the maintenance problems and increased shielding requirements that would result from the plate-out of entrained radioactivity in the low-temperature components. The primary system was designed with two separate loops, each delivering heat to its own intermediate heat exchanger, to provide redundancy in decay heat removal in the event of a malfunction or damage to the system. Lithium was elected as the reactor coolant because its high specific heat and low vapor pressure would permit much smaller pipe sizes and wall thickness and lower pumping power. Existing irradiation test data related to fuel dimensional stability, fuel/clad/coolant compatibility data, and thermophysical properties led to the choice of uranium nitride as the prime nuclear fuel material. (Recent irradiation data developed under the Liquid-Metal

Fast Breeder Reactor (LMFBR) program indicate that fuel swelling of uranium carbide, one of the candidate NuERA fuel materials, would not be as severe as had been anticipated in the NuERA study.)

The reactor concept resulting from the evaluation of viable design options was a lithium-cooled compact fast reactor. All the primary piping material was a columbium alloy (Cb-1Zr), as were the intermediate heat exchangers, the pumps, and the reactor vessel. The fuel pins, which also were in contact with the lithium, were clad in a tantalum-base alloy (ASTAR 811-C). A secondary fluid, liquid sodium-potassium (NaK), was used to transfer the reactor thermal energy to the engines. It exited the shell of the intermediate heat exchanger at 1700°F and flowed to the engine heat exchanger. In this engine counterflow exchanger, the NaK was cooled to 1300°F as its energy was transferred to the air stream, thereby replacing the normal combustor function in the engine. Isolation valves were included in each secondary loop to provide shutoff capability if required in the event of an engine failure.

As shown in Figure 3, the reactor with its primary shield was mounted vertically with the vertical centerline laterally offset aft from the center of a spherical containment vessel. The intermediate heat exchangers, the primary pumps, and the shield coolant system heat exchanger and pumps were installed in the crescent-shaped space provided. The banana-shaped expansion tank was located at the highest point in the compartment and was connected to one of the reactor outlet lines. The intermediate heat exchangers, primary coolant pumps, and expansion tank were supported from the containment shell. The reactor and shield assembly were mounted on a base structure attached to the lower portion of the containment shell, which was designed to withstand a 200-ft/s impact in any direction without rupturing.

The primary shield completely enclosed the reactor and consisted of an inner layer of tungsten inside the pressure vessel, an external layer of zirconium hydride, and an outer layer of lithium hydride. Inserted between these two major layers was a sheet of boral. The shield was compartmented and hermetically sealed to prevent the outgassing of materials and the need to contain the shield coolant.

The decay heat removal system made use of one of the primary and secondary loops. The latter was modified to contain a bypass loop that was connected to one of the secondary loops and rejected the decay heat through a liquid metal to air heat exchanger. The secondary coolant pump provided the pumping power.

The NuERA conceptual design developed during this extended study satisfied all reactor performance criteria and complied with specified size and weight constraints on the nuclear subsystem. Thus the basic feasibility of using nuclear power for manned aircraft propulsion using state-of-the-art reactor technology was established.

Innovative Aircraft Design Study (IADS) (Muehlbauer, 1977). An
extension of the NuERA study was subsequently performed for the U.S
Air Force by Lockheed-Georgia Company and Westinghouse in 1976-1977 to
estimate the lightest ramp weight of an aircraft configuration with a
nuclear propulsion system. Parametric analyses and design-refinement
studies were conducted for conventional, canard, and spanloader
aircraft configurations with payloads of 400,000 and 600,000 lb of
containerized and/or outsized cargo, a cruise Mach number of 0.75, an
emergency range of 1,000 n. mi (nautical miles) on chemical fuel, and
a field length of 9,000 ft. The canard configuration was selected for
detailed analysis, since it had the lowest ramp weights of the three
configurations for both payloads.

Analyses of several Rankine and Brayton nuclear propulsion cycles
resulted in the selection of an open Brayton turbofan cycle for a
reference aircraft. The selection was made on the basis of the
extensive data background and low weight of the cycle. Of all the
cycles considered, only a nonrecuperated closed Brayton cycle with a
dual-mode engine was found to be lighter in weight than the selected
cycle. However, this closed Brayton cycle was excluded from
consideration in the IADS study because an adequate data base was
lacking.

The reference aircraft design conforming with mission requirements
had a ramp weight of 1,556,000 lb and served as a basis for assessing
the sensitivity of the design to variations in the mission
requirements, in advanced technology applications, and in the nuclear
operation and design philosophy. A 13.1 percent reduction in ramp
weight to 1,353,000 lb was achieved by adopting an alternate nuclear
operational design philosophy. Features of this alternative included
special shaping of the shield, use of the emergency range chemical
fuel for shielding, and full-power reactor operation for all normal
flight phases with half-power for emergencies. The use of composite
materials for 40 percent of the structural weight of the reference
aircraft produced 13.5 percent savings in aircraft ramp weight
relative to all-aluminum aircraft.

Nuclear Bi-Brayton System (Pierce, 1979). Early Westinghouse studies
of nuclear aircraft propulsion concepts focused on liquid-metal-cooled
fast reactors (NuERA) using the open Brayton cycle and a
high-temperature intermediate heat exchanger, which presented design
problems and weight penalties. Subsequent investigations of
alternative nuclear propulsion systems indicated that a Bi-Brayton
cycle utilizing an advanced High-Temperature Gas-Cooled Reactor was a
promising concept.

The gas-cooled reactor selected for this concept made use of
already proven technologies, primarily from the NERVA nuclear rocket
program and from commercial gas-cooled reactors. The reactor was a
helium-cooled, graphite-moderated epithermal reactor with TRISO-coated
fuel beads dispersed in extruded graphite elements. It had a lateral

support system to maintain core bundling and a beryllium radial reflector with control drums.

The Bi-Brayton cycle eliminated the need for high-temperature capabilities in the engine and intermediate heat exhangers and permitted the coupling of a state-of-the-art gas-cooled reactor. Shielding criteria and materials for the reactor were the same as those for the NuERA System. The arrangement of reactor components within the containment vessel is shown in Figure 4. Figure 5 contains a system schematic and reveals how the physical separation of the primary and secondary energy transport loops was achieved; system temperatures are also shown, including those for the neat exchangers, which are substantially lower than those in a comparable NuERA system. Another advantage of these low temperatures was that concentric stainless steel supply and return lines could be used for the helium secondary system, thereby reducing the piping weight and eliminating the need for insulation.

Table 1 reveals the power plant weight reductions obtainable with a Bi-Brayton system in comparison with those of the liquid-metal concept using the closed Brayton cycle.

Nuclear Marine Propulsion (Thompson, 1975; Jones, 1975)

Extensive studies were performed by Westinghouse in applying state-of-the-art reactor technology to evolving requirements for lightweight power and propulsion systems. One such study was initiated in 1972 to investigate the feasibility of a ship powered by a lightweight nuclear propulsion (LWNP) system. Key assumptions used in this study included a ship weight of 2,000 long tons, a maximum mission duration of 6 months, a power plant design lifetime of 10,000 equivalent full-power hours, and a power plant specific weight less than 15 lb per shaft horsepower (SHP) at 140,000 horsepower. Design requirements included compliance with federal regulations on radiological safety (10 CFR 50) and containment integrity following a 30-kn (knot) collision.

Several reactor options and power conversion options were evaluated in the early phase of this effort. Following appropriate trade-off studies, a NERVA-type reactor and a closed-cycle helium gas-turbine power conversion system were selected for this application. Although direct shaft power could have been employed, superconducting motors and generators were chosen to permit transfer of large blocks of power to numerous locations throughout the ship. A schematic diagram of this power plant is shown in Figure 6. Only one of the two identical parallel power conversion loops is shown. Reactor outlet conditions of $1700^{\circ}F$ and 1,500 psia were selected after consideration of cycle and materials requirements and technology. These conditions provided an attractive overall cycle efficiency and a compact power plant that can be achieved within the expected capability of superalloy materials without requiring cooling of the turbine blades.

TABLE 1 Comparison of System Weights for Liquid-Metal (NuERA) and
Gas-Cooled (Bi-Brayton) Nuclear Aircraft Propulsion Systems

| | Subsystem Weights (1,000 lb) | |
	Liquid-Metal NuERA	Gas-Cooled Bi-Brayton
Nuclear subsystem	391.3	419.3
NSS auxiliaries	15.0	15.0
Piping and engine HX	122.7	32.3
Turbine and gearing	--	6.4
Engines	81.0	70.3
JP fuel system	3.1	3.1
JP fuel	134.6	122.1
Total system weight	747.7	669.5

The nuclear reactor, along with the two power conversion packages,
radiation shielding, and containment, is shown in Figure 7. This unit
has a 140,000-SHP rating and is 32 ft long, 18 ft wide, and 34 ft
high. The auxiliaries, not shown in this figure, are estimated to
require 11,700-ft^3 additional volume. Tne entire primary system is
surrounded by a thick-walled containment vessel and consists of two
cylinders joined in the form of an inverted "T." The 2-in.-thick
vertical cylinder immediately surrounds the reactor assembly, while
the lower 5-in.-thick cylinder surrounds the turbomachinery, emergency
cooling system, control gas storage bottles, and power conversion
equipment. Tnis vessel, in addition to providing for mounting of
system components, acts as tne third level of containment for fission
products, provides for protection of tne system in the event of
collision, and satisfies a portion of the total system shielding
requirements. The number of connections through tnis vessel is
minimized and includes means for positive sealing to ensure protection
against release of fission products in the event of an accident.

The reactor design drew heavily upon the demonstrated NERVA reactor
design technology but incorporated adaptations of those NERVA design
features appropriate for this LWNP application. Thus on the basis of
the successful NERVA development and test program, there was a high

degree of confidence that the LWNP reactor could be successfully designed and built with a minimum of research and development.

Like NERVA, the reactor consisted of a gas-cooled, graphite-moderated, epithermal core with coated fuel beads dispersed in graphite elements; it had a lateral support system to maintain core bundling and a beryllium radial reflector with control drums. While fuel element fabrication was based on NERVA technology, the lower operating temperature permitted the use of TRISO-design fuel beads developed and used in commercial gas-cooled reactors. This feature enhanced retention of fission products within the fuel bead itself and substantially increased the overall safety of the system.

The specific weight of the power plant design described above for a 140,000-SHP rating was less than 12 lb/SHP, well within the 15-lb/SHP specification. The power plant components comprising this total weight included all components, controls, and auxiliaries up to the connection to the waterjets or propeller shafts and the lift-fan shafts. The study revealed that the power plant could also be designed with an arrangement whereby the turbomachinery and heat exchangers would be separated and more accessible, but not without weight and size penalties that were undesirable for the LWNP application.

The development of the LWNP conceptual design described above, including the evaluation and application of available technology, indicated that a lightweight nuclear propulsion power plant was feasible and could be developed with minimal risk and reasonable R&D costs.

Space Electric Power (Thompson and Pierce, 1977)

The purpose of this internal Westinghouse study was to investigate the feasibility of designing a spacecraft nuclear power plant with output in excess of 50 kW(e) that would have a high degree of commonality with various space reactor applications in order to minimize development cost. The concept adopted for this application consisted of a gas-cooled, graphite-moderated reactor with a direct Brayton-cycle power conversion unit. An inert gas mixture was heated in the reactor and delivered to the power conversion units through the center of a coaxial piping arrangement. The power conversion units contained the turbine-compressor-alternator rotating unit, a recuperator, a gas management system, an electric control system, and a heat rejection system.

This concept accommodated the restrictive and unique requirements associated with space applications, including reliability, performance, development risk, and cost. Many investigators have assessed the closed Brayton cycle anbd confirmed its inherent advantages for space power applications. The Brayton power conversion turbomachinery has growth potential; since system power level is approximately proportional to pressure level, a single set of

turbomachinery can be developed for use over a wide range of power output. The NASA-Lewis Research Center and the AiResearch Manufacturing Co. have demonstrated a closed Brayton rotating unit with impressive lifetime and reliability. Although the concept developed in this study utilized two parallel 25-kW(e) Brayton units, additional units sized for fractional or full system power could be incorporated at relatively small weight penalties to enhance system reliability.

A gas-cooled reactor is a natural heat source for the closed Brayton cycle, and the use of NERVA technology for this concept took advantage of the advanced development status of this lightweight, high-power-density reactor. Like NERVA, the reactor consisted of a gas-cooled, graphite-moderated, epitnermal core with coated fuel beads dispersed in graphite elements. While fuel element fabrication was based on NERVA technology, the lower operating temperature permitted the use of TRISO-design fuel beads, which enhanced the retention of fission products and reduced the potential level of fission product contamination of the working fluid. In view of the relatively low core exit temperature, the reactor incorporated a not-end support plate developed as an alternate core support design to the in-core tie-rods used in NERVA. This was a significant design feature in minimizing the size of the core and external shield.

This NERVA-type reactor is sized by end-of-life criticality requirements such that the reactor core size is essentially the same for any power rating less than 600 kW(e). Therefore the system becomes particularly attractive from a weight standpoint at higher power ratings.

Total weight of the 50-kW(e) space application utilizing the graphite-moderated, gas-cooled reactor with Brayton power conversion units was estimated to be 5,300 lb, as given in Table 2. The Brayton power conversion component weights were scaled from AiResearch Manufacturing Company data. The radiator weight estimate was based on 22 ft^2/kW for a total radiator area of 1,100 ft^2. The radiation shield consisted of 1,120 lb of canned lithium hydride in the form of a shadow cone and was sized for a lifetime neutron dose of 10^{15} nvt to protect an unmanned electronic payload. A system layout for a rocket launch vehicle is shown in Figure 8.

This in-house study revealed that a closed-cycle Brayton system using a gas-cooled reactor would have a high degree of commonality in technology and development for a variety of missions and would be a feasible system for space power application. The study also indicated that such a power plant would be highly competitive with other types of nuclear systems at power levels as low as 100 kW(e); even at power levels as low as 50 kW(e), this type of nuclear system appeared to be viable.

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TABLE 2 Graphite-Moderated, Gas-Cooled Space Power Plant (50 kW(e)-50,000 EFPH): Weight Summary

	Weight, lb
Reactor	
Core including filler strips	415
Beryllium reflector and lateral support	505
Inlet support and reflector	531
Exit support, reflector, and shielding	650
Pressure vessel	392
Actuators	52
Subtotal	2,545
Power Conversion System	
Turbomachinery (2 at 25 kW(e))	145
Recuperators	132
Radiator	1,100
Controls and start-up equipment	80
Radiator and reactor ducts	185
Subtotal	1,642
Additional shielding	1,120
Total Weight	5,307

SUMMARY AND CONCLUSIONS

The NERVA program demonstrated the feasibility of high-power-density, gas-cooled reactors for space applications and left a broad technology legacy for future power and propulsion programs. NuERA and IADS work verified the technical and operational viability of using a high-temperature, liquid-metal-cooled reactor for manned aircraft propulsion. The follow-on Nuclear Bi-Brayton System study revealed that an aircraft propulsion system using a gas-cooled reactor would have attractive design and weight advantages. The LWNP study established the merits of applying NERVA technology to an integrated power system in order to meet stringent weight and safety constraints. Finally, the space electric power study, based on gas-cooled reactor technology, established the flexibility and overall superiority of nuclear reactors for space power applications at levels as low as 100 kW(e). Numerous proprietary studies on l weight gas-cooled and liquid-metal reactor power and propulsion syste have been performed by Westinghouse since the programs described above were completed and have generally validated these results.

On the basis of this extensive participation in the design, development, and testing of lightweight, high-temperature reactor power systems, we believe that (1) the feasibility of using gas-cooled or liquid-metal-cooled reactors for aerospace power and propulsion applications has been firmly established and (2) the existing technology base justifies high confidence that an aerospace reactor development program could meet Department of Defense requirements on performance, cost, and schedule if the program is initiated promptly with a firm and clearly defined mission.

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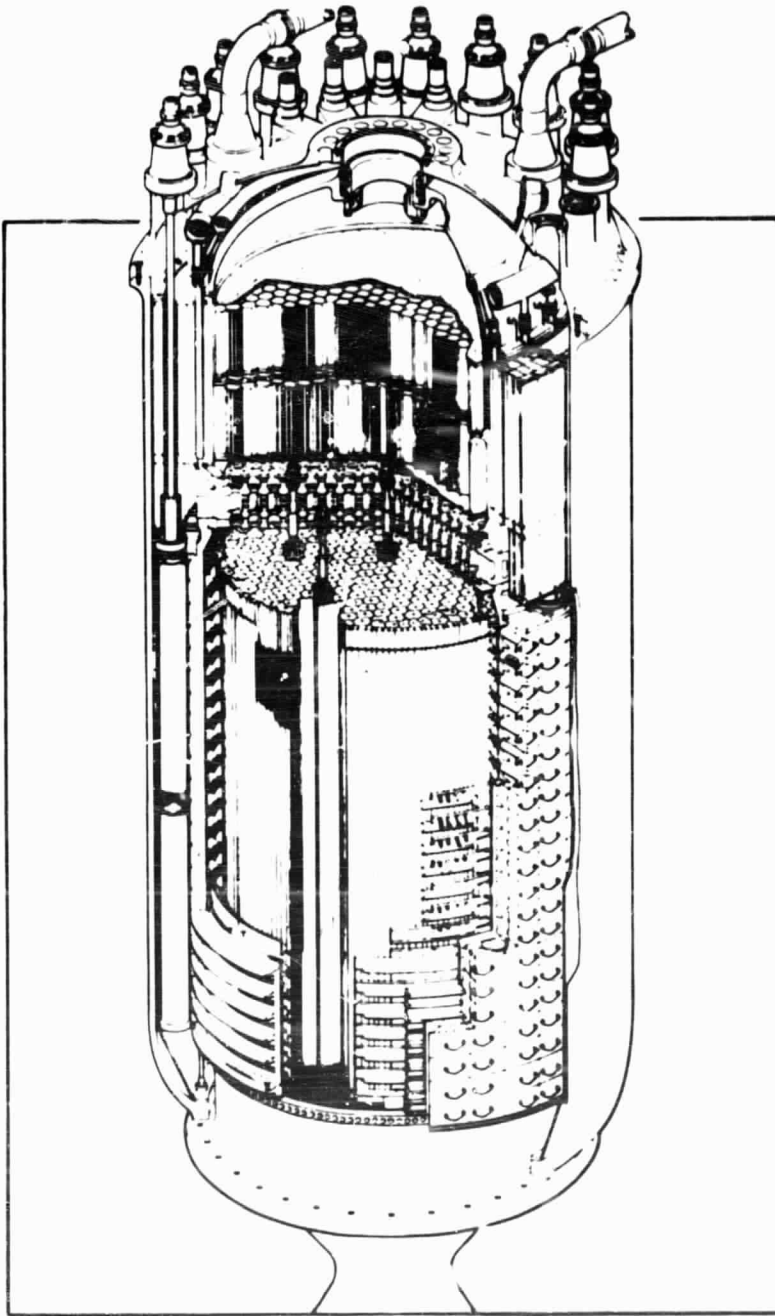
FIGURE 1   NERVA reactor assembly.
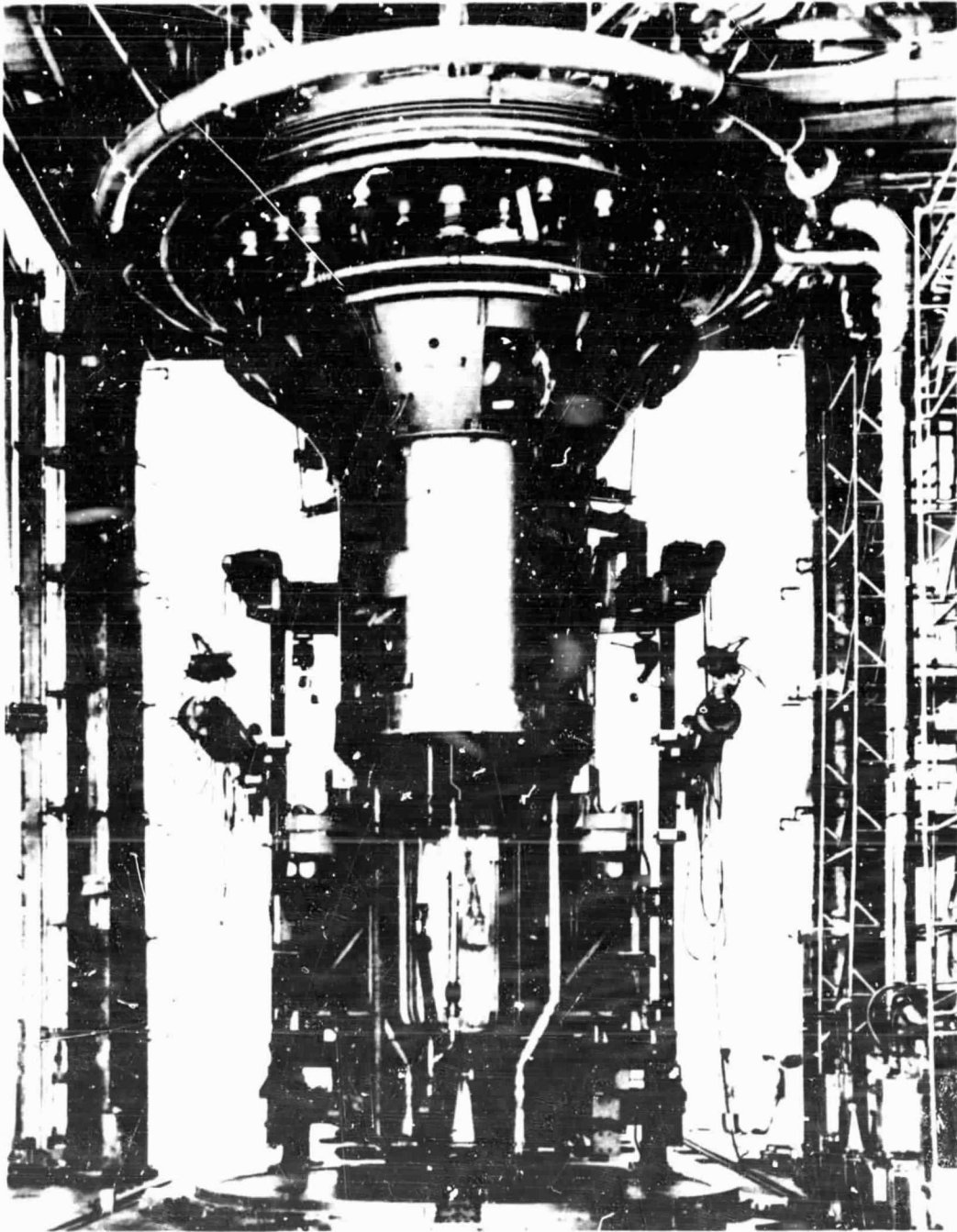
FIGURE 2 NERVA-XE engine at test stand.

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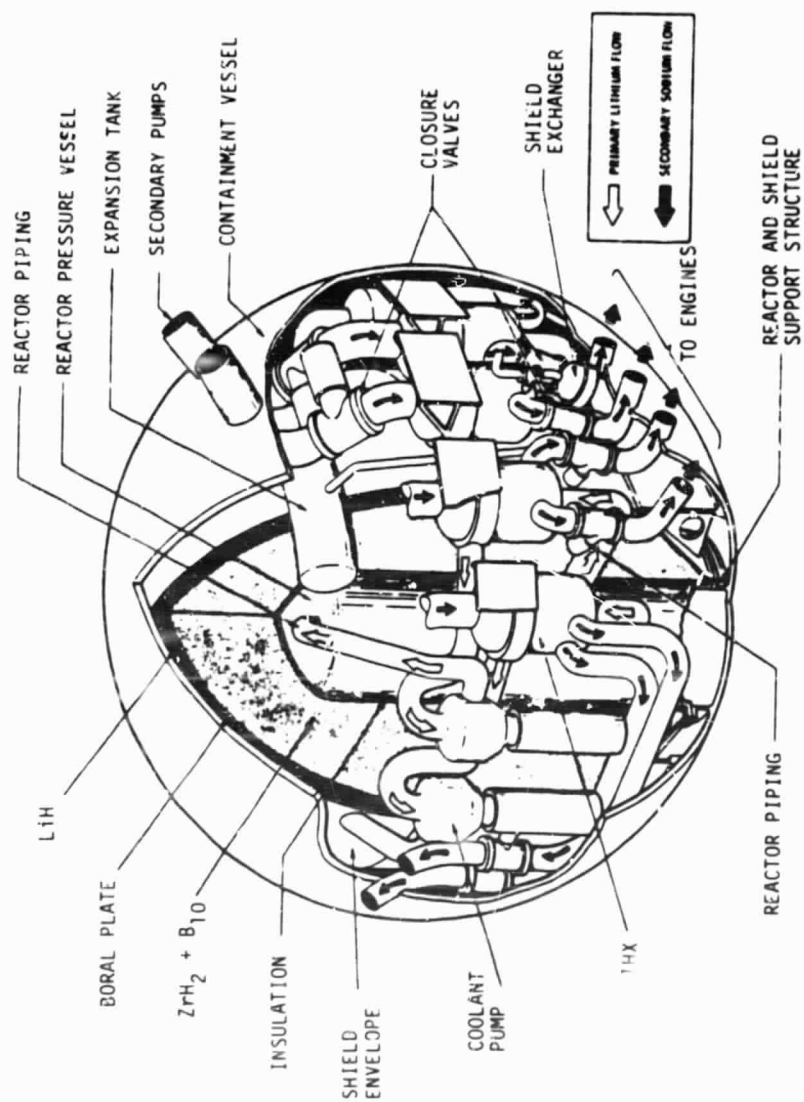
FIGURE 3 NuERA (liquid metal) nuclear subsystem for aircraft propulsion.
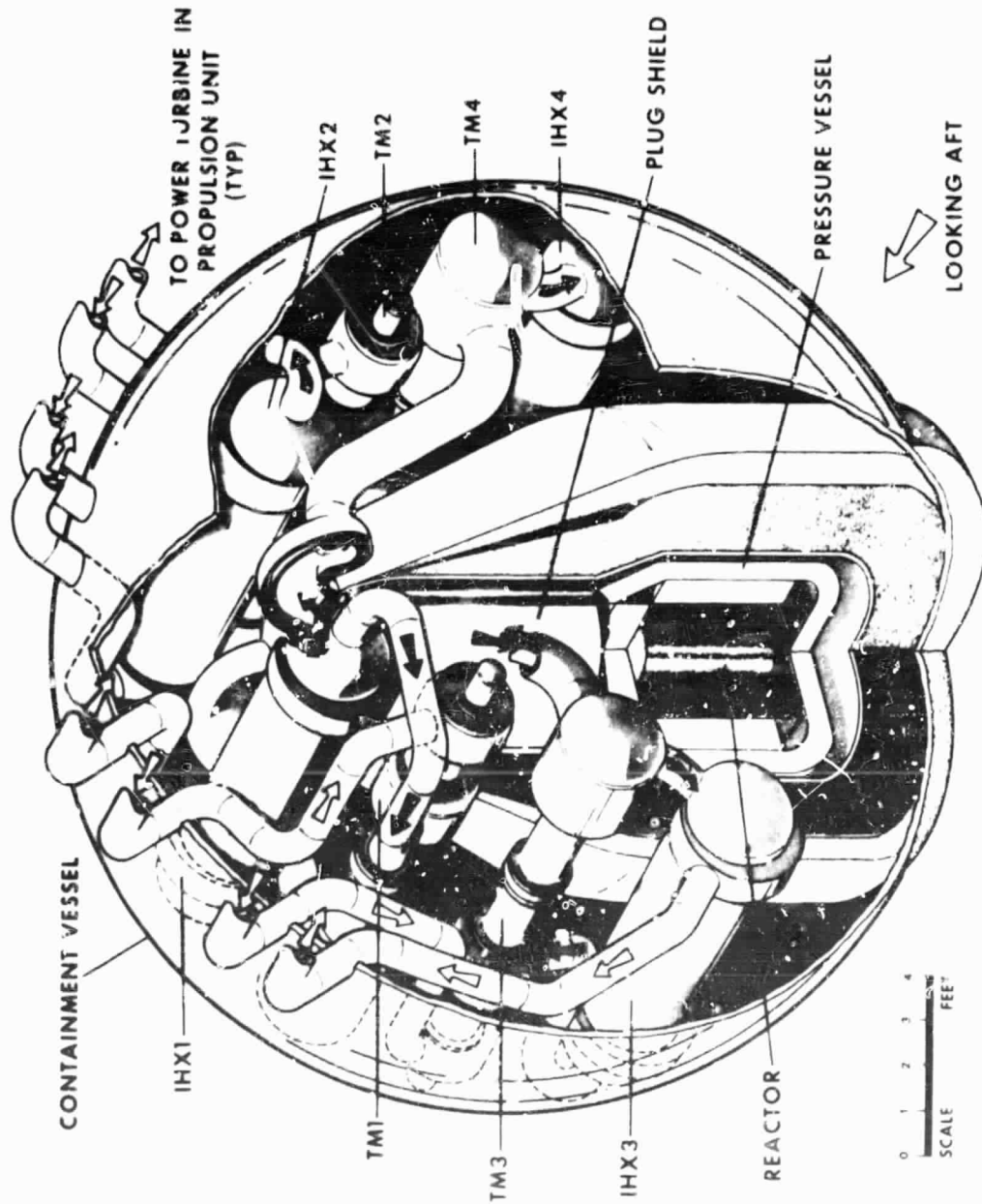
FIGURE 4 Bi-Brayton (gas-cooled) nuclear subsystem for aircraft propulsion.
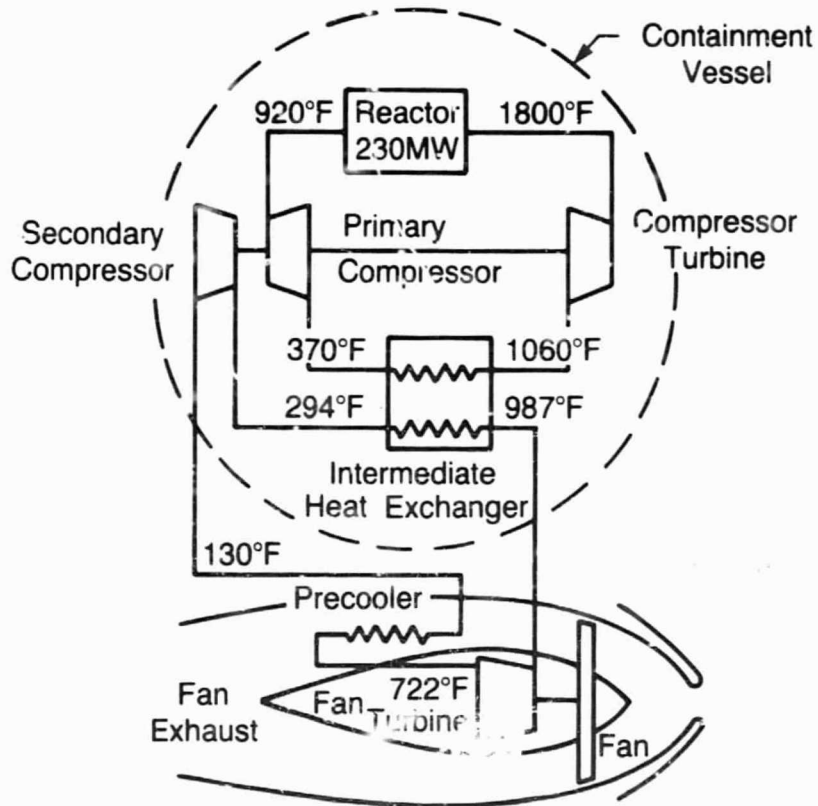
FIGURE 5   Schematic diagram of Bi-Brayton aircraft propulsion system.
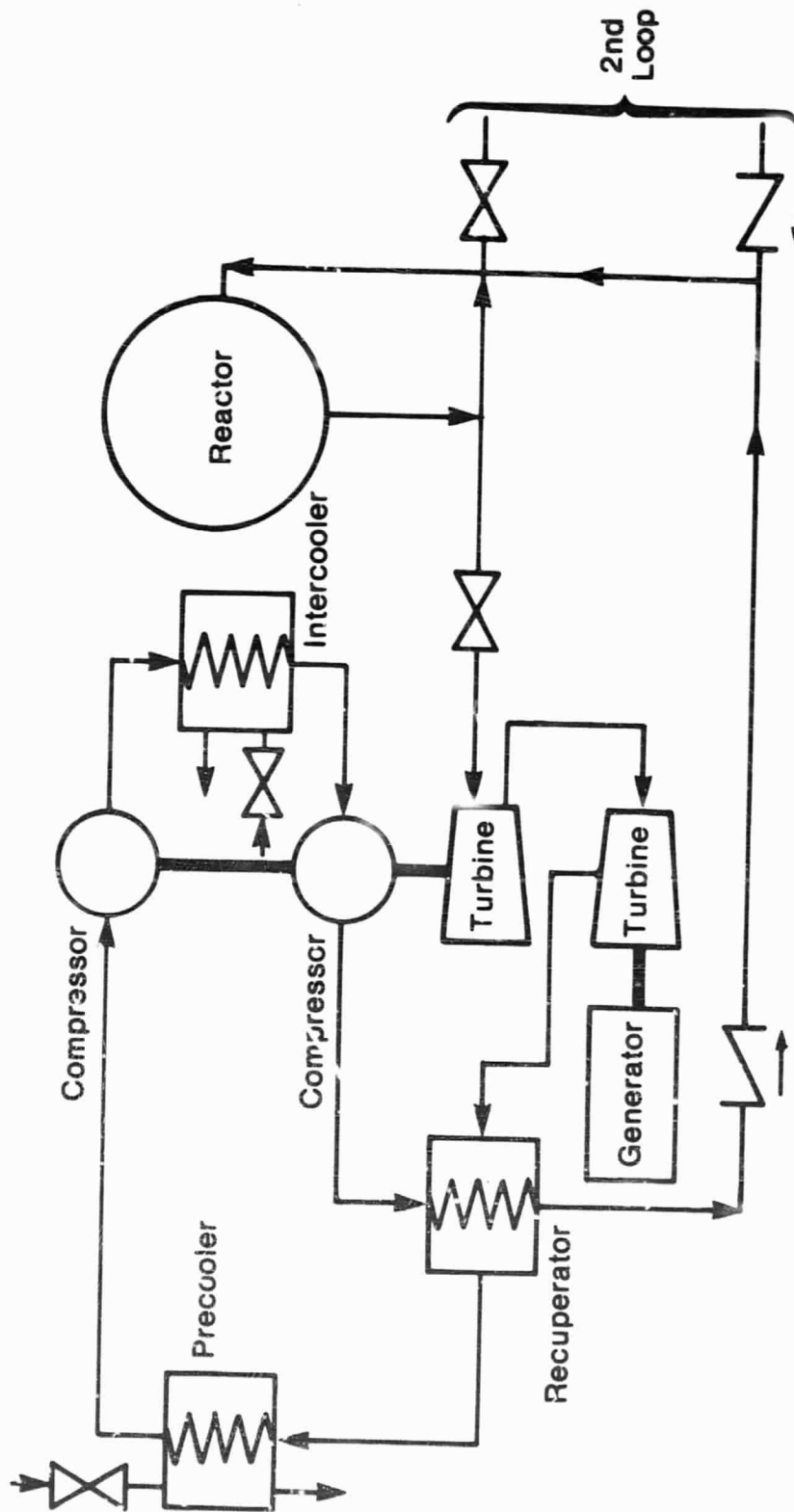
FIGURE 6   Schematic diagram of LWNP nuclear power plant.

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PRIMARY SHIELD
. TUNGSTEN
. ZIRCONIUM HYDRIDE
. WATER

PRESSURE VESSEL

CORE (FUEL)

CONTROL DRUM IN REFLECTOR

CONTAINMENT

EMERGENCY COOLING

TURBINE - COMPRESSOR - HEAT EXCHANGE PACKAGE

SUPERCONDUCTING GENERATOR

UPPER SUPPORT (HORIZONTAL)

CONTROL DRUM DRIVE

INLET END REFLECTOR (HOLD DOWN)

PLUG SHIELD

SYSTEM MAIN SUPPORT

HELIUM BOTTLE

SUPPORT LINK
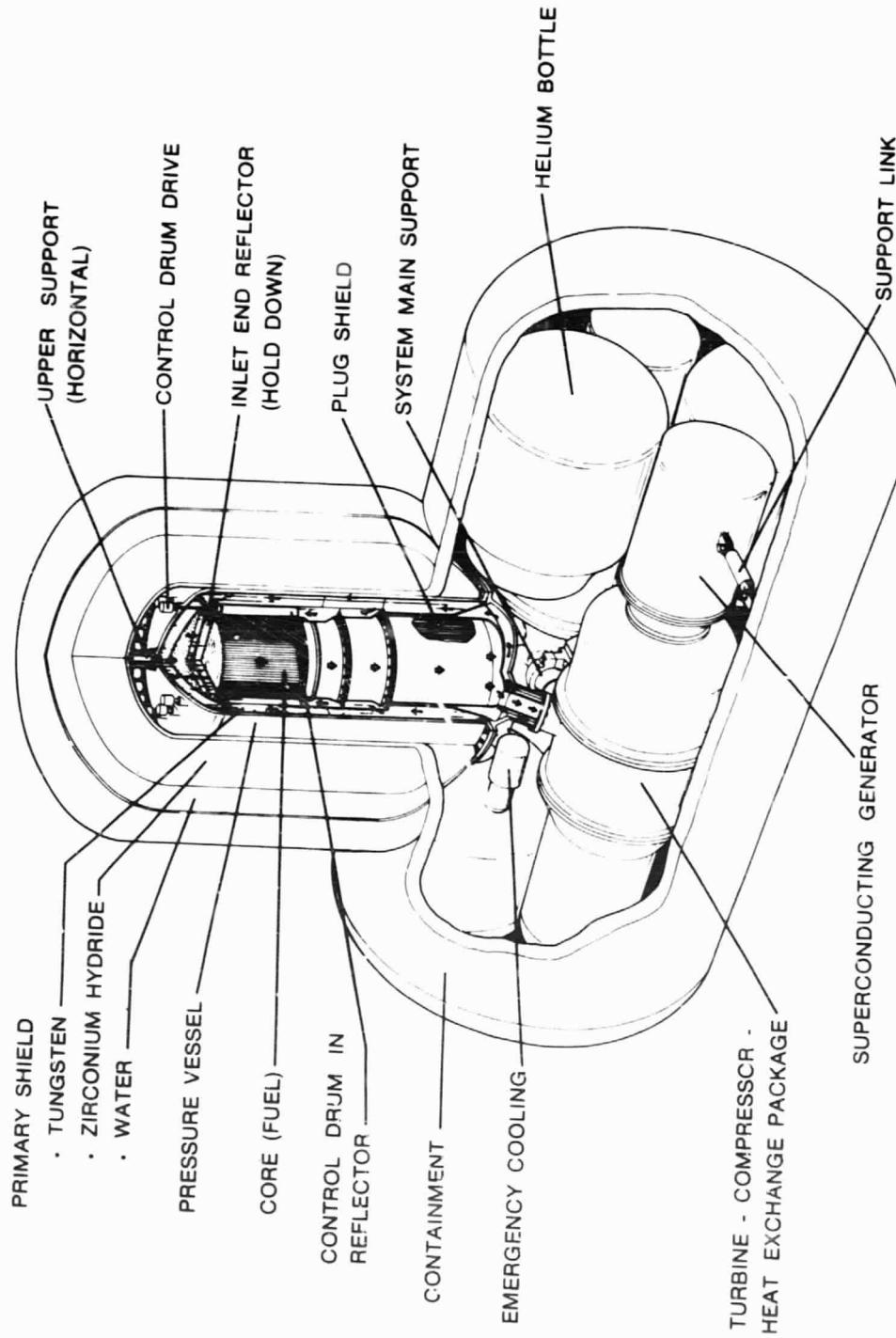
FIGURE 7 Layout of LWNP power plant.
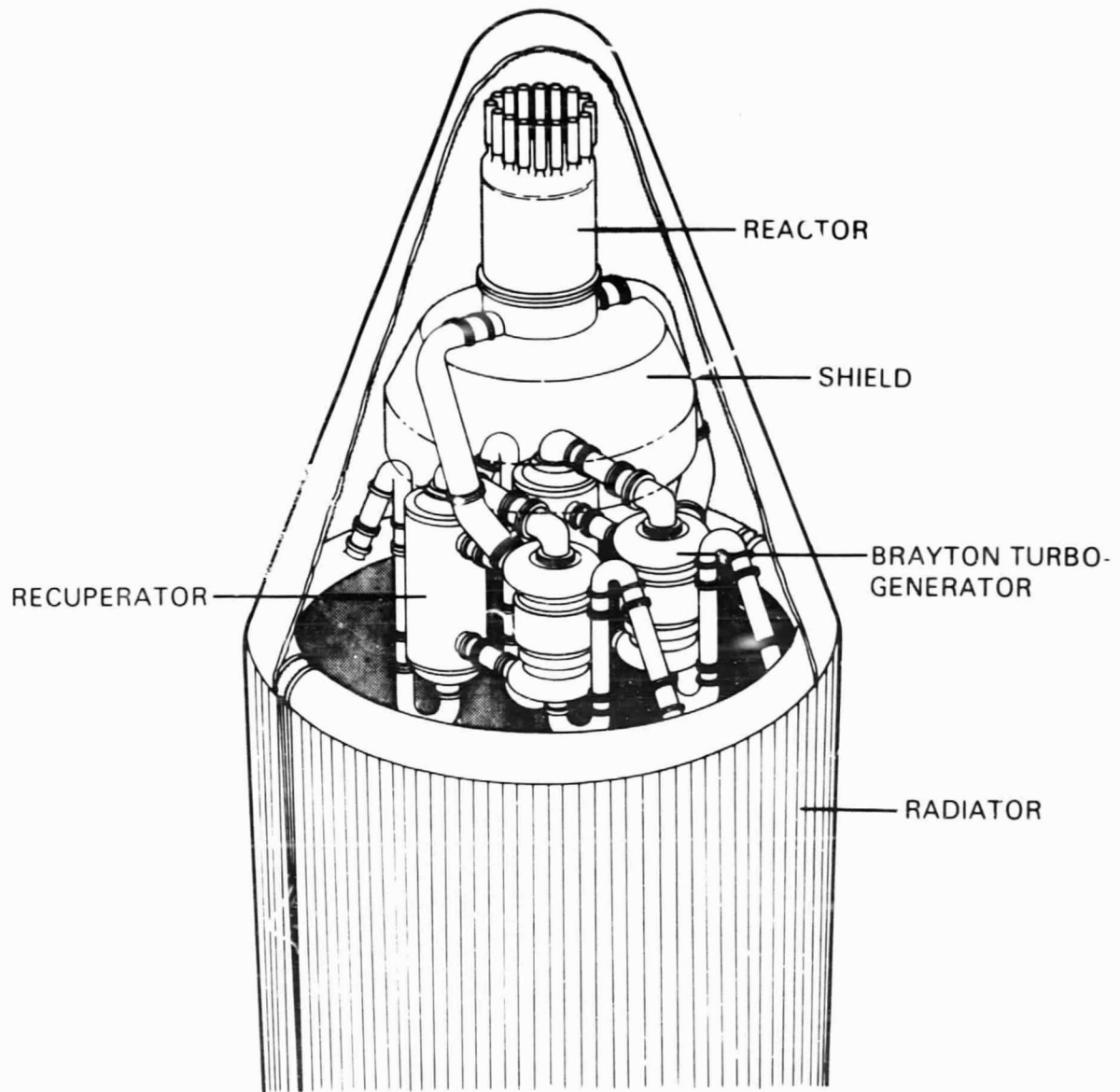
FIGURE 8 Layout of nuclear Brayton-cycle space power plant.

TECHNOLOGICAL IMPLICATIONS OF SNAP REACTOR
POWER SYSTEM DEVELOPMENT FOR FUTURE SPACE
NUCLEAR POWER SYSTEMS ACTIVITIES

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ABSTRACT

Although the Space Transportation System brings a new era to U.S. space
technology, the United States still lags, the Soviet Union in total space
launches per year and in the development of nuclear systems for higher
electric powers in space. The power levels for future U.S. military and
civilian spa missions are not fully defined at this time; however,
power needs can easily be projected to much higher values than are
required today. Nuclear reactor systems are one method of satisfying
these power needs. The development of such systems must proceed on a
path consistent with mission needs and schedules. This path, or
technology road map, starts from the power system technology data base
available today. Much of this data base was established during the 1960s
and early 1970s, when government and industry developed space nuclear
reactor systems for steady state power and propulsion. One of the
largest development programs was the Systems for Nuclear Auxiliary Power
(SNAP) program. By the early 1970s, a technology base had evolved from
this program at the system, subsystem, and component levels, with many
implications on future reactor power systems. A review of this
technology base highlights the need for a power system technology and
mission overview study. Such a study is currently being performed by
Rockwell's Energy Systems Group for the Department of Energy and will
assess power system capabilities versus mission needs, considering
development, schedule, and cost implications. The end product of the
study will be a technology road map to guide reactor power system
development.

INTRODUCTION

In the 1980s, with the advent of the Space Shuttle, a new era in U.S.
space technology began. The Space Shuttle is ultimately capable of
lifting 30-t (tonne) payloads into low earth orbits (Figure 1). Not only
is the shuttle a reusable launch vehicle, it is also a viable space
experimental laboratory. Using sortie missions, experiments can

be launched, carried out in the environment of space, and recovered (Figure 2). The total space transportation system (STS) will eventually encompass upper stages for placement of missions into higher altitude and geosynchronous equatorial orbits (Figure 3).

However, in the 1980s, there also occurred a change in perspective regarding the role of the United States as military and technology leader of the world. Even with the advancements in launch capability, our chief political competitor, the Soviet Union, has "out launched" the United States: 88 launches versus 12 in the first 10 months of 1982. Of the Soviet Union's 88 launches, 71 were for military purposes, compared to only 3 of the United States' 12 launches.

On the other hand, the U.S. leadership in military technology became apparent during the Lebanon war of 1982. This war focused on Western war-fighting technology and demonstrated the superiority of this technology over that supplied by the Soviet Union to its allies. The conclusion is that USSR capabilities tend to exceed U.S. capabilities in quantity but not in quality.

One technology area in which the Soviet Union has been successful in both quantity and quality is the development of space nuclear reactor power systems. Seventeen years ago, the United States launched a single space reactor power system called the SNAP-10A (Figure 4). This system, launched in a spacecraft called SNAPSHOT in 1965, was a flight qualification system developed for the Atomic Energy Commission and the Air Force at Atomics International (now a division of Rockwell's Energy Systems Group). It is interesting to note that the original USSR space reactor power system, called "ROMASHKA," was a derivative of a very early static SNAP-10 concept (Figure 5). Since the SNAPSHOT launch in 1965, the Soviet Union has improved the quality of its space nuclear reactor power capabilities through a thermionic system called "TOPAZ." A derivative of the ROMASHKA and TOPAZ systems provides power for its radar ocean reconnaissance satellites (RORSATs) (Figure 6). Approximately 20 RORSATs have been launched by the Soviet Union, with four launches in the first 10 months of 1982. Although the RORSAT nuclear power system is neither high powered nor long lived, the Soviet Union's use of reactors exemplifies its belief that power requirements for space will increase over time.

For the United States, these increasing power requirements will come from military missions (surveillance and war-fighting) and also from civilian missions (scientific and large commercial). The real power system needs for these missions can only be estimated at this time. To assess these needs, two parameters in addition to power must be evaluated: the duration of the power and the mass of the system providing the power. The product of the power and its duration is the energy output of the system, and both the power and the energy affect the system mass. The mass in turn affects the total launch requirements for the mission. For the types of missions being discussed by military and civilian planners today, ranges in power, power duration, and power system mass can be defined and

interrelated. Power requirements range over 4 decades, from kW(e) to 100 MW(e). Power duration requirements range over 6 decades, from seconds of semicontinuous power for directed energy applications to near 100,000 hours for long-lived, steady state applications. System mass requirements range over 3 decades, from hundreds of kilograms to tens of thousands of kilograms. When plotted in a three-dimensional log-log-log coordinate system, the interrelation of these three parameters defines a "requirements space" (Figure 7). When examining this space, the difference between steady state and semicontinuous needs becomes apparent. What does not become apparent is the level of power system technology necessary to meet the real needs of military and civilian applications. The schedules and budgets required to develop power systems that are time compatible with missions must be factored into a plan for development--a technology road map. The road map for power systems must begin with the complete nuclear power system technology base that exists in the United States today.

BACKGROUND

The U.S. development of a steady state space nuclear reactor power system technology base began in the early 1950s. Reactor systems were code named with even-numbered "SNAP" designators and were initially considered with power capabilities from hundreds of watts to multikilowatts. The principal SNAP reactor power systems were SNAP-2, SNAP-8, SNAP-10A, SNAP-50, and the SNAP follow-on concepts (Figure 8). All the numerically designated systems were designed for a 1-year power duration, with power capabilities ranging from 0.5 to 350 kW(e). The follow-on system concepts were aimed at 2- to 7-year lives with power capabilities up to 75 kW(e). The SNAP-2, -8, -10A, and follow-on systems were based on a thermal reactor using uranium-zirconium hydride fuel. The SNAP-50 was based on a fast reactor using uranium nitride fuel.

The hardware phase of the SNAP program occurred in the 1960s and centered around the SNAP-2, -8, -10A, and -50 systems. Separate reactor and power conversion subsystem development and testing were performed for the SNAP-2 and -8 systems. The SNAP-10A system included reactor development, power conversion subsystem development, and fully operational system testing on the ground and in space. The SNAP-50 program included component development testing and did not proceed through reactor or system fabrication and testing. The SNAP follow-on concepts were oriented at the system technology improvement, and their data base was drawn largely from the SNAP-2, -8, and -10A programs. The major hardware milestones in the SNAP program included the SNAP-10A ground test and flight tests in 1965 and the SNAP-8 reactor demonstration tests in the 1960s (Figure 9).

Paralleling this hardware phase was the Aerospace Nuclear Safety Program (Figure 9). Approximately $50 million (1960s dollars) was

spent in this program alone to verify the safety of nuclear reactor systems before launch, during launch, in space, and upon reentry.

Early in the 1970s, system concepting and subsystem improvement activities were under way. These activities included the design of various reactor power systems, such as the 5-kW(e) reactor thermoelectric system and the 75-kW(e) reactor turboelectric system (Figure 9). At the end of this improvement phase, the uranium-zirconium hydride reactor system technology base was well on its way to providing a 25-kW(e) system with a 5-year life. Over the range of powers being considered at that time, specific powers of 6-30 W/kg were projected. These systems had longer lifetimes and specific energy capabilities of 30-210 W-yr/kg. The uranium nitride reactor system technology base was oriented to much higher powers, with SNAP-50 type systems offering large increases in specific power and energy.

The SNAP program was terminated in the late 1970s. At that time, several hundred million dollars had been invested in the technology base, with over $200 million of this investment at Atomics International.

SYSTEM, SUBSYSTEM, AND COMPONENT IMPLICATIONS

The technology base that was most completely demonstrated and supported by test resulted from the SNAP-10A and -8 programs. This base was also used in the SNAP follow-on programs. Another technology base was established by the higher-power, higher-temperature SNAP-50 program.

The SNAP-10A, -8, and follow-on programs considered various types of power conversion, but the reactor and shielding concepts were similar for all systems. The reactors were thermal, based on uranium-zirconium hydride fuel. The hydrogen moderator was intrinsically bound in the fuel matrix to minimize fuel inventory and maximize reactor safety. Fuel composition for all reactors was generally the same, using 10 wt% (weight percent) fully enriched uranium and 90 wt% zirconium. The fuel was hydrided to 6×10^{22} nydrogen atoms/cm^3, or about the same hydrogen density as water. The reactors were cooled with a liquid-metal eutectic, NaK-78 (78 wt% potassium). Reactor neutron shielding used lithium hydride (LiH) in a stainless steel containment vessel. Gamma shielding, when employed, used heavy metals.

The SNAP-10A system, launched on April 3, 1965, was designated Flight System 4 (Figure 10). The total spacecraft was called SNAPSHOT and included the Agena vehicle with primary and secondary payloads. The SNAPSHOT spacecraft was launched into a 1,300-km (700-n. mi. (nautical mile)) orbit with approximately a 3,500-year orbit life. The SNAP-10A reactor power system initially produced 580 W(e) in orbit. SNAP-10A was designed to be actively controlled only at the beginning of its life; after the reactor was stabilized, electric

power drifted down with time. SNAP-10A performance in space tracked the performance of the ground test system (Flight System 3, see below). At the forty-third day of in-space operation, following orbit 553, telemetry from the SNAPSHOT spacecraft was lost. It was later determined that thermal overstresssing of the Agena voltage regulator caused an erroneous shutdown command to the reactor's permanent shutdown device.

The SNAP-10A system had a single NaK loop and employed silicon-germanium thermoelectric elements (Figure 11). The reactor coolant outlet temperature was 833 K (1040°F). The unshielded reactor mass per unit thermal energy was 2,650 kg/MW(t)-yr. The overall system had a specific power of 1.3 W/kg. Because the system was designed for a 1-year life, the specific energy was equal to the specific power.

SNAP-10A Flight System 3 was tested on the ground in vacuum for 10,000 hours at power and temperature (Figure 12). The ground test was initiated prior to Flight System 4 launch so that flight performance could be tracked against ground performance. The ground test system operated at the same reactor outlet temperature and average radiator temperature as the flight test system. The effective higher sink temperature of the ground facility versus that of the space facility resulted in lower power output from the ground test system.

Because of the concerns of launching a reactor system into space, a significant safety program supported the SNAPSHOT launch (Figure 13). This program was called the Aerospace Nuclear Safety Program and supported reactor and radioisotope system development activities. Both safety analysis and testing were performed for the SNAPSHOT launch. In the Reentry Flight Demonstration Test, a full-scale nonfueled and nonradioactive replica of the SNAP-10A reactor was launched and subjected to a suborbital flight path. This test demonstrated beryllium reflector separation from the reactor vessel and also supported theoretical modeling in areas such as aerodynamic heating and reactor disassembly.

A highlight of the safety program was a series of reactor transient tests called the SNAPTRAN experiments. These experiments were used to measure the response to rapid reactivity insertions and core water flooding. SNAPTRAN-3 was specifically conducted to characterize the SNAP-10A reactor in the water-flooding event that could follow a launch abort (Figure 14). In this experiment, a SNAP-10A core was located in a water-filled environmental chamber. The core was surrounded by a poisoned and voided sleeve. The sleeve was pyrotechnically removed and the reactor was destructed. The expansion of the well-instrumented core was followed by three high-speed cameras, and the data collected were used to determine coefficients of reactivity versus volume expansion. Experimental data also verified that the reactivity temperature coefficient was -0.4¢/°C prior to the initiation of core expansion. The SNAPTRAN-3 experiment further demonstrated that fuel element disintegration was caused by the

generation of hydrogen overpressure. The one intact fuel element from the experiment snowed that this hydrogen evolution was nonuniform about the cross section of the fuel (Figure 14). A very positive outcome of the experiment was confirmation that a high fraction of the fission products were retained in the fuel element.

The SNAP-8 program was directed at a 600-kW(t) reactor with a 978 K (1300°F) coolant outlet temperature. The reactor was to be used with higher-electric-power mercury Rankine turboelectric power conversion subsystems. Two complete reactors were tested in this program: the SNAP-8 Experimental Reactor (S8ER) and the SNAP-8 Developmental Reactor (S8DR).

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(Brayton and Rankine) power conversion subsystems. Total system specific powers were projected at 6-30 W/kg, depending on the absolute power level. Total system specific energies were projected at 30-210 W-yr/kg.

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successfully demonstrated in the SNAP program. These components used metallic bellows to accommodate the differential volume expansion between the liquid-metal coolant and the piping. The volume compensators also provided pressurization for the liquid-metal coolant, while maintaining it void free. Fifty volume compensators were tested at elevated temperatures for over 100,000 total hours. The SNAP-10A ground system compensator accumulated 10,000 test hours.

The power conversion and processing subsystem data base includes technologies for thermoelectric and turboelectric power conversion (Figure 21). Although the SNAP-10A system used silicon-germanium thermoelectrics, development efforts were also focused on lead telluride (PbTe) because of its higher potential figure of merit, Z (equals material Seebeck coefficient squared divided by electrical resistivity and thermal conductivity). Lead telluride compact tubular thermoelectric converters were designed by Westinghouse Astronuclear Laboratory and tested by both Westinghouse and Atomics International. Two hundred and twenty units were tested for 720,000 hours, with a single unit operating for 42,000 hours. A figure of merit of 0.8 x $10^{-3}K^{-1}$ was demonstrated for these converters, i.e., an efficiency of 5 percent at a hot clad temperature of 858 K (1085°F) and a cold clad temperature of 537 K (507°/F), the required system temperatures. The turboelectric subsystems included the mercury Rankine units for SNAP-2 and -8 that were developed by TRW and Aerojet, respectively. A mercury Rankine subsystem was tested in a full-up SNAP-2 reactor power system configuration using an electrically simulated reactor. Thirty thousand hours of test experience were demonstrated on the mercury Rankine units with a 5,000-hour single-duration test.

In reviewing the reactor power system technology data base that evolved from the SNAP program, the question may be asked, "What does it mean today?" Over the past 10 years, no further activities were performed on these compact reactor systems. However, in support of the radioisotope thermoelectric generator (RTG) programs, development activities did proceed with silicon-germanium thermoelectric power conversion.

An interesting perspective can be gained by "concepting" a SNAP-type reactor power system using a SNAP-evolved U-ZrH reactor with current silicon-germanium thermoelectric capabilities (Figure 22). To be consistent with currently envisioned mission requirements, a power system life of 7 years is used. As the design electrical power level of this system concept is increased, the specific power and specific energy capabilities of the system increase dramatically. Although SNAP-10A exhibited a specific power and energy of only 1 3 W/kg, a SNAP-evolved derivative system with a 100-kW(e) capability exhibits a specific power of 11 /kg and specific energy of 80 W-yr/kg. Such large improvements are due to the economy of size, to increased reactor coolant temperature (from 833 to 922 K), and to improved thermoelectric converter efficiency. This system concept would still be very large and heavy in high-power ranges. The important

conclusion is the dramatic system-level effect of power on the mass of such a system.

A further perspective into this effect can be gained by looking at the same type of SNAP-evolved power system concept, but one utilizing improved thermoelectrics (Figure 23). These improved thermoelectrics have a figure of merit of $1.0 \times 10^{-3} K^{-1}$ and have been proposed as a reasonable technology improvement. The specific power achievable by such a system concept at 100 kW(e) is 15 W/kg. The specific energy achievable at this power is over 100 W-yr/kg.

A review and an evaluation of the technology data base established by the Pratt and Whitney SNAP-50 reactor turboelectric system provide a similarly dramatic demonstration of the system-level effect of power on mass. SNAP-50 was being configured for power levels of 300-1000 kW(e). The system employed a fast uranium nitride reactor, operating at a lithium coolant temperature of 1367 K (2000°F). The reactor was projected to have an unshielded mass per unit energy of 730 kg/MW(t)-yr. The projected system conversion efficiency was 14 percent using a potassium Rankine turboelectric power conversion subsystem. SNAP-50 was being designed for a 1-year life and was projected to have an unshielded specific power in the range of 100-200 W/kg.

The key conclusion from this review and evaluation of the SNAP technology data base is that an "order-of-magnitude" power system technology improvement is not required to meet the specific power and specific energy goals envisioned for future space missions. These goals are currently in the ranges of 40-50 W/kg and 250-350 W-yr/kg. Extrapolations of all types of past system technologies demonstrate that these goals seem reasonable with newer, more advanced technologies. It is only necessary to determine the level of technology advancement necessary to meet real mission needs

MISSION CONSIDERATIONS

Reactor power systems have always been directed at next-generation, far-term missions. In the early 1960s, reactor-powered space station concepts were being studied (Figure 24). As power system concepts and missions evolved, the 1970s envisioned reactor-powered lunar and space stations (Figure 25). In the 1980s, the Galileo mission to explore Jupiter is being planned (Figure 26).

Today, the most important activity for the space nuclear power community is determining if any mission requires a reactor power system. As a starting point in the search for a mission, we can note the Soviet Union's use of reactor power for RORSAT. Surveillance is a potential mission with a potential near-term need. A possible technology for this mission is the synthetic aperture radar, which had its initial demonstration in the shuttle imaging radar (SIR-A) experiment. SIR-A was built by the Jet Propulsion Laboratory as part of the OSTA-1 (Office of Space and Terrestrial Applications)

experimental package (Figure 27). SIR-A operated at an altitude of 260 km (140 n. mi.). Interesting views from SIR-A include an oil slick near Venezuela (Figure 28), oil derricks near Abu Dhabi in the Persian Gulf (Figure 29), and freeways in the Los Angeles basin (Figure 30). SIR-A required a power of only 1 kW(e). It does take much imagination to envision the capabilities of a near-term radar system having 20 or 30 kW(e) of power.

A radar surveillance system is only one potential mission. It does, however illustrate one truism--mission power needs will increase with time, and power system capabilities must follow those needs. As the power needs continue to increase, the necessity for reactor power systems will increase. Looking again at the mission "requirements space" (from Figure 7) with a time variable imposed, one can see a possible progression of power system capabilities over time (Figure 31). This time dependence is different for steady state and semicontinuous missions. In the late 1980s, there could be near-term missions, and only certain power system technologies may be capable of meeting near-term mission needs. In the early 1990s and mid-1990s, there could be midterm and far-term missions. Midterm and far-term reactor power systems will be available to meet midterm and far-term mission needs only if proper planning is performed now. This planning is the technology road map that fits time-phased mission needs to realistically achievable power systems.

TECHNOLOGY-MISSION OVERVIEW

The determination of such a technology road map was recommended to the Department of Energy by Rockwell International. The Energy Systems Group of Rockwell International is currently performing a technology-mission overview study to determine this road map for the Department of Energy. The study uses a "bottom-up" approach to assess power system capabilities and ties these capabilities to validated mission needs generated by the Department of Defense and the National Aeronautics and Space Administration (Figure 32).

An important feature of the study is technology assessment. In the "bottom-up" study approach, reactor power system technologies are first assessed at the subsystem level. Five subsystems are defined for this assessment (Figure 33). For each subsystem, current technologies are assesed, as are the future capabilities of improved technologies for midterm needs and advanced technologies for far-term needs. Of prime importance is the consideration of development time and development cost for each improved or advanced technology. When complete, the technology assessment provides time-compatible building blocks, (i.e., subsystems whose development can be achieved in the same time frame. These building blocks are then used to generate systems whose schedular availability can be time phased by near-term, midterm, and far-term initial operating capability (IOC).
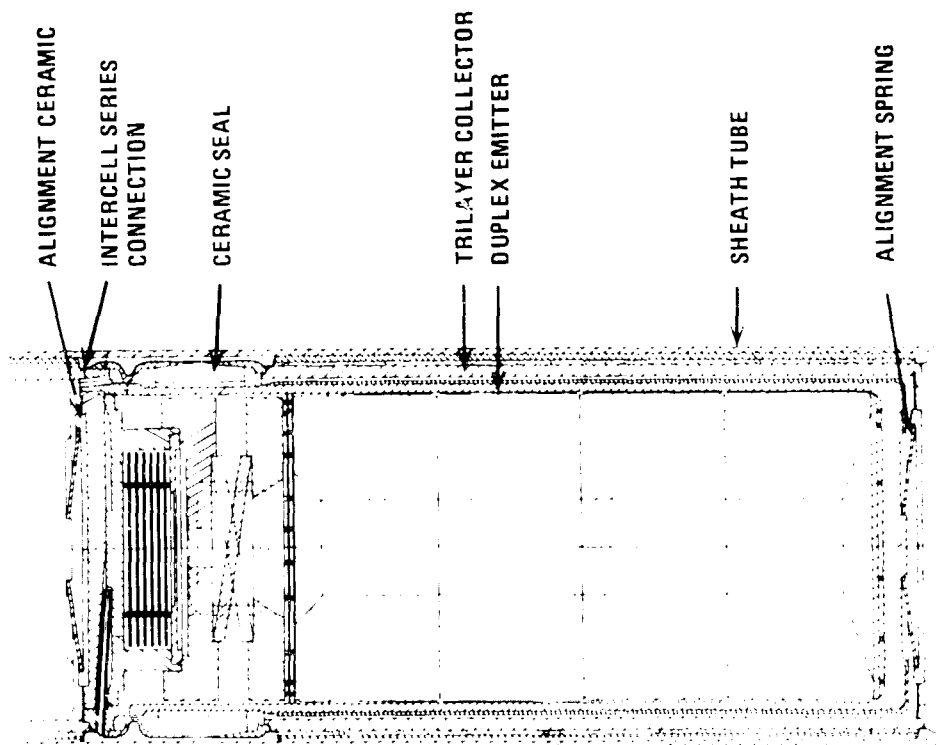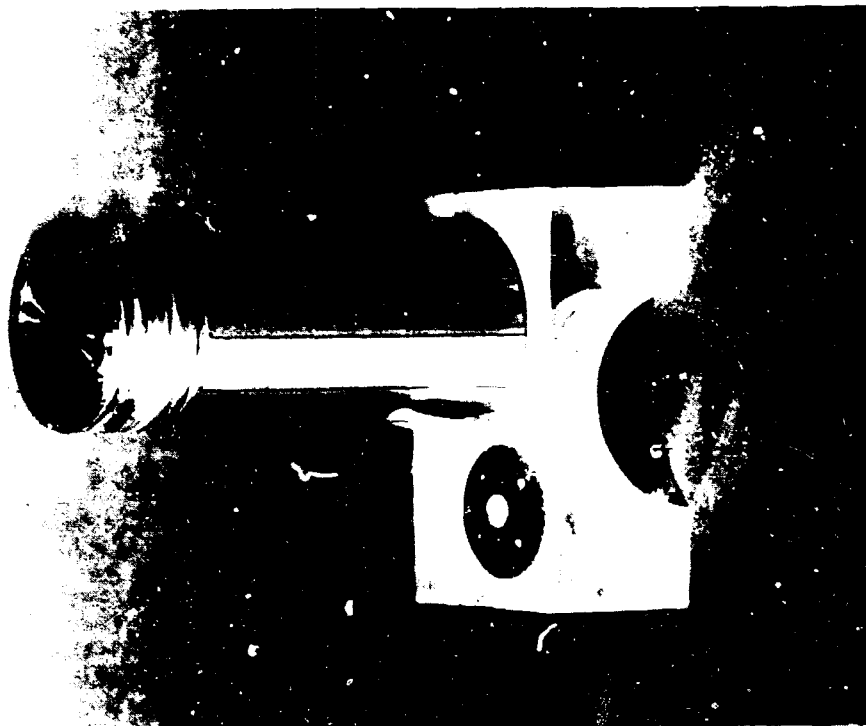
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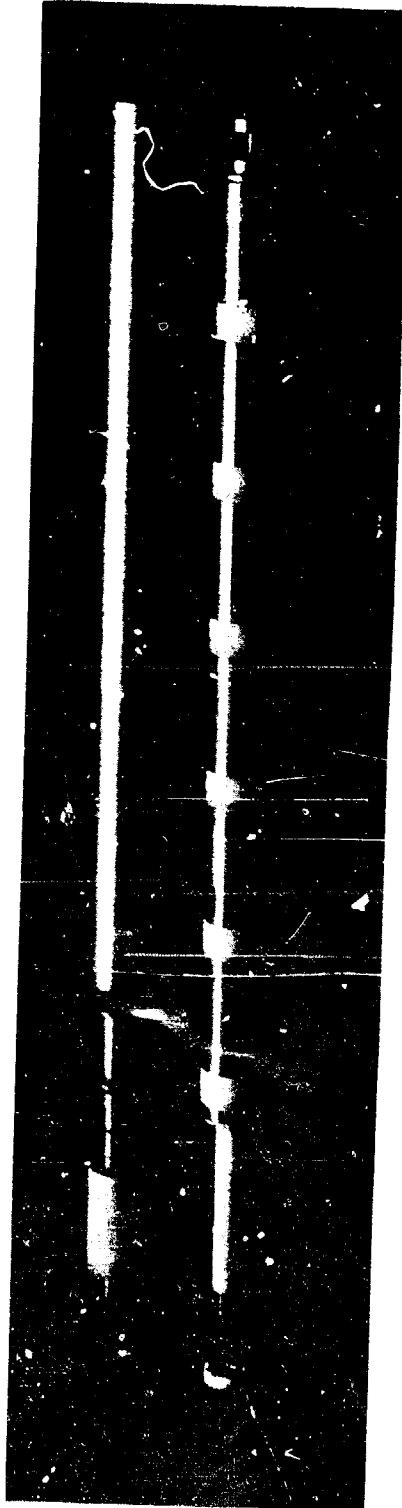SHEATH TUBE

ALIGNMENT SPRING

FIGURE 10 Thermionic cell.

FIGURE 11 Thermionic fuel element (TFE).

TECHNOLOGICAL IMPLICATIONS OF SNAP REACTOR POWER SYSTEM DEVELOPMENT FOR FUTURE SPACE NUCLEAR POWER SYSTEMS ACTIVITIES

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ABSTRACT

Although the Space Transportation System brings a new era to U.S. space technology, the United States still lags, the Soviet Union in total space launches per year and in the development of nuclear systems for higher electric powers in space. The power levels for future U.S. military and civilian space missions are not fully defined at this time; however, power needs can easily be projected to much higher values than are required today. Nuclear reactor systems are one method of satisfying these power needs. The development of such systems must proceed on a path consistent with mission needs and schedules. This path, or technology road map, starts from the power system technology data base available today. Much of this data base was established during the 1960s and early 1970s, when government and industry developed space nuclear reactor systems for steady state power and propulsion. One of the largest development programs was the Systems for Nuclear Auxiliary Power (SNAP) program. By the early 1970s, a technology base had evolved from this program at the system, subsystem, and component levels, with many implications on future reactor power systems. A review of this technology base highlights the need for a power system technology and mission overview study. Such a study is currently being performed by Rockwell's Energy Systems Group for the Department of Energy and will assess power system capabilities versus mission needs, considering development, schedule, and cost implications. The end product of the study will be a technology road map to guide reactor power system development.

INTRODUCTION

In the 1980s, with the advent of the Space Shuttle, a new era in U.S. space technology began. The Space Shuttle is ultimately capable of lifting 30-t (tonne) payloads into low earth orbits (Figure 1). Not only is the shuttle a reusable launch vehicle, it is also a viable space experimental laboratory. Using sortie missions, experiments can

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be launched, carried out in the environment of space, and recovered (Figure 2). The total space transportation system (STS) will eventually encompass upper stages for placement of missions into higher altitude and geosynchronous equatorial orbits (Figure 3).

However, in the 1980s, there also occurred a change in perspective regarding the role of the United States as military and technology leader of the world. Even with the advancements in launch capability, our chief political competitor, the Soviet Union, has "out launched" the United States: 88 launches versus 12 in the first 10 months of 1982. Of the Soviet Union's 88 launches, 71 were for military purposes, compared to only 3 of the United States' 12 launches.

On the other hand, the U.S. leadership in military technology became apparent during the Lebanon war of 1982. This war focused on Western war-fighting technology and demonstrated the superiority of this technology over that supplied by the Soviet Union to its allies. The conclusion is that USSR capabilities tend to exceed U.S. capabilities in quantity but not in quality.

One technology area in which the Soviet Union has been successful in both quantity and quality is the development of space nuclear reactor power systems. Seventeen years ago, the United States launched a single space reactor power system called the SNAP-10A (Figure 4). This system, launched in a spacecraft called SNAPSHOT in 1965, was a flight qualification system developed for the Atomic Energy Commission and the Air Force at Atomics International (now a division of Rockwell's Energy Systems Group). It is interesting to note that the original USSR space reactor power system, called "ROMASHKA," was a derivative of a very early static SNAP-10 concept (Figure 5). Since the SNAPSHOT launch in 1965, the Soviet Union has improved the quality of its space nuclear reactor power capabilities through a thermionic system called "TOPAZ." A derivative of the ROMASHKA and TOPAZ systems provides power for its radar ocean reconnaissance satellites (RORSATs) (Figure 6). Approximately 20 RORSATs have been launched by the Soviet Union, with four launches in the first 10 months of 1982. Although the RORSAT nuclear power system is neither high powered nor long lived, the Soviet Union's use of reactors exemplifies its belief that power requirements for space will increase over time.

For the United States, these increasing power requirements will come from military missions (surveillance and war-fighting) and also from civilian missions (scientific and large commercial). The real power system needs for these missions can only be estimated at this time. To assess these needs, two parameters in addition to power must be evaluated: the duration of the power and the mass of the system providing the power. The product of the power and its duration is the energy output of the system, and both the power and the energy affect the system mass. The mass in turn affects the total launch requirements for the mission. For the types of missions being discussed by military and civilian planners today, ranges in power, power duration, and power system mass can be defined and

interrelated. Power requirements range over 4 decades, from kW(e) to 100 MW(e). Power duration requirements range over 6 decades, from seconds of semicontinuous power for directed energy applications to near 100,000 hours for long-lived, steady state applications. System mass requirements range over 3 decades, from hundreds of kilograms to tens of thousands of kilograms. When plotted in a three-dimensional log-log-log coordinate system, the interrelation of these three parameters defines a "requirements space" (Figure 7). When examining this space, the difference between steady state and semicontinuous needs becomes apparent. What does not become apparent is the level of power system technology necessary to meet the real needs of military and civilian applications. The schedules and budgets required to develop power systems that are time compatible with missions must be factored into a plan for development—a technology road map. The road map for power systems must begin with the complete nuclear power system technology base that exists in the United States today.

BACKGROUND

The U.S. development of a steady state space nuclear reactor power system technology base began in the early 1950s. Reactor systems were code named with even-numbered "SNAP" designators and were initially considered with power capabilities from hundreds of watts to multikilowatts. The principal SNAP reactor power systems were SNAP-2, SNAP-8, SNAP-10A, SNAP-50, and the SNAP follow-on concepts (Figure 8). All the numerically designated systems were designed for a 1-year power duration, with power capabilities ranging from 0.5 to 350 kW(e). The follow-on system concepts were aimed at 2- to 7-year lives with power capabilities up to 75 kW(e). The SNAP-2, -8, -10A, and follow-on systems were based on a thermal reactor using uranium-zirconium hydride fuel. The SNAP-50 was based on a fast reactor using uranium nitride fuel.

The hardware phase of the SNAP program occurred in the 1960s and centered around the SNAP-2, -8, -10A, and -50 systems. Separate reactor and power conversion subsystem development and testing were performed for the SNAP-2 and -8 systems. The SNAP-10A system included reactor development, power conversion subsystem development, and fully operational system testing on the ground and in space. The SNAP-50 program included component development testing and did not proceed through reactor or system fabrication and testing. The SNAP follow-on concepts were oriented at system technology improvement, and their data base was drawn largely from the SNAP-2, -8, and -10A programs. The major hardware milestones in the SNAP program included the SNAP-10A ground test and flight tests in 1965 and the SNAP-8 reactor demonstration tests in the 1960s (Figure 9).

Paralleling this hardware phase was the Aerospace Nuclear Safety Program (Figure 9). Approximately $50 million (1960s dollars) was

spent in this program alone to verify the safety of nuclear reactor systems before launch, during launch, in space, and upon reentry.

Early in the 1970s, system concepting and subsystem improvement activities were under way. These activities included the design of various reactor power systems, such as the 5-kW(e) reactor thermoelectric system and the 75-kW(e) reactor turboelectric system (Figure 9). At the end of this improvement phase, the uranium-zirconium hydride reactor system technology base was well on its way to providing a 25-kW(e) system with a 5-year life. Over the range of powers being considered at that time, specific powers of 6-30 W/kg were projected. These systems had longer lifetimes and specific energy capabilities of 30-210 W-yr/kg. The uranium nitride reactor system technology base was oriented to much higher powers, with SNAP-50 type systems offering large increases in specific power and energy.

The SNAP program was terminated in the late 1970s. At that time, several hundred million dollars had been invested in the technology base, with over $200 million of this investment at Atomics International.

SYSTEM, SUBSYSTEM, AND COMPONENT IMPLICATIONS

The technology base that was most completely demonstrated and supported by test resulted from the SNAP-10A and -8 programs. This base was also used in the SNAP follow-on programs. Another technology base was established by the higher-power, higher-temperature SNAP-50 program.

The SNAP-10A, -8, and follow-on programs considered various types of power conversion, but the reactor and shielding concepts were similar for all systems. The reactors were thermal, based on uranium-zirconium hydride fuel. The hydrogen moderator was intrinsically bound in the fuel matrix to minimize fuel inventory and maximize reactor safety. Fuel composition for all reactors was generally the same, using 10 wt% (weight percent) fully enriched uranium and 90 wt% zirconium. The fuel was hydrided to 6×10^{22} nydrogen atoms/cm^3, or about the same hydrogen density as water. The reactors were cooled with a liquid-metal eutectic, NaK-78 (78 wt% potassium). Reactor neutron shielding used lithium hydride (LiH) in a stainless steel containment vessel. Gamma shielding, when employed, used heavy metals.

The SNAP-10A system, launched on April 3, 1965, was designated Flight System 4 (Figure 10). The total spacecraft was called SNAPSHOT and included the Agena vehicle with primary and secondary payloads. The SNAPSHOT spacecraft was launched into a 1,300-km (700-n. mi. (nautical mile)) orbit with approximately a 3,500-year orbit life. The SNAP-10A reactor power system initially produced 580 W(e) in orbit. SNAP-10A was designed to be actively controlled only at the beginning of its life; after the reactor was stabilized, electric

power drifted down with time. SNAP-10A performance in space tracked
the performance of the ground test system (Flight System 3, see
below). At the forty-third day of in-space operation, following orbit
553, telemetry from the SNAPSHOT spacecraft was lost. It was later
determined that thermal overstresssing of the Agena voltage regulator
caused an erroneous shutdown command to the reactor's permanent
shutdown device.

The SNAP-10A system had a single NaK loop and employed
silicon-germanium thermoelectric elements (Figure 11). The reactor
coolant outlet temperature was 833 K (1040°F). The unshielded
reactor mass per unit thermal energy was 2,650 kg/MW(t)-yr. The
overall system had a specific power of 1.3 W/kg. Because the system
was designed for a 1-year life, the specific energy was equal to the
specific power.

SNAP-10A Flight System 3 was tested on the ground in vacuum for
10,000 hours at power and temperature (Figure 12). The ground test
was initiated prior to Flight System 4 launch so that flight
performance could be tracked against ground performance. The ground
test system operated at the same reactor outlet temperature and
average radiator temperature as the flight test system. The effective
higher sink temperature of the ground facility versus that of the
space facility resulted in lower power output from the ground test
system.

Because of the concerns of launching a reactor system into space, a
significant safety program supported the SNAPSHOT launch (Figure 13).
This program was called the Aerospace Nuclear Safety Program and
supported reactor and radioisotope system development activities.
Both safety analysis and testing were performed for the SNAPSHOT
launch. In the Reentry Flight Demonstration Test, a full-scale
nonfueled and nonradioactive replica of the SNAP-10A reactor was
launched and subjected to a suborbital flight path. This test
demonstrated beryllium reflector separation from the reactor vessel
and also supported theoretical modeling in areas such as aerodynamic
heating and reactor disassembly.

A highlight of the safety program was a series of reactor transient
tests called the SNAPTRAN experiments. These experiments were used to
measure the response to rapid reactivity insertions and core water
flooding. SNAPTRAN-3 was specifically conducted to characterize the
SNAP-10A reactor in the water-flooding event that could follow a
launch abort (Figure 14). In this experiment, a SNAP-10A core was
located in a water-filled environmental chamber. The core was
surrounded by a poisoned and voided sleeve. The sleeve was
pyrotechnically removed and the reactor was destructed. The expansion
of the well-instrumented core was followed by three high-speed
cameras, and the data collected were used to determine coefficients of
reactivity versus volume expansion. Experimental data also verified
that the reactivity temperature coefficient was $-0.4\cent/^{\circ}$C prior to
the initiation of core expansion. The SNAPTRAN-3 experiment further
demonstrated that fuel element disintegration was caused by the

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The primary heat transport subsystem data base encompasses piping, liquid-metal, electromagnetic pump, and coolant volume compensator technologies (Figure 20). In the SNAP program, 58 electromagnetic pumps were tested for 212,000 total hours, with a single pump operating successfully for 42,000 hours. These pumps operated at temperatures up to 920 K (1200°F). Volume compensators were also

conclusion is the dramatic system-level effect of power on the mass of such a system.

A further perspective into this effect can be gained by looking at the same type of SNAP-evolved power system concept, but one utilizing improved thermoelectrics (Figure 23). These improved thermoelectrics have a figure of merit of $1.0 \times 10^{-3} K^{-1}$ and have been proposed as a reasonable technology improvement. The specific power achievable by such a system concept at 100 kW(e) is 15 W/kg. The specific energy achievable at this power is over 100 W-yr/kg.

A review and an evaluation of the technology data base established by the Pratt and Whitney SNAP-50 reactor turboelectric system provide a similarly dramatic demonstration of the system-level effect of power on mass. SNAP-50 was being configured for power levels of 300-1000 kW(e). The system employed a fast uranium nitride reactor, operating at a lithium coolant temperature of 1367 K (2000°F). The reactor was projected to have an unshielded mass per unit energy of 730 kg/MW(t)-yr. The projected system conversion efficiency was 14 percent using a potassium Rankine turboelectric power conversion subsystem. SNAP-50 was being designed for a 1-year life and was projected to have an unshielded specific power in the range of 100-200 W/kg.

The key conclusion from this review and evaluation of the SNAP technology data base is that an "order-of-magnitude" power system technology improvement is not required to meet the specific power and specific energy goals envisioned for future space missions. These goals are currently in the ranges of 40-50 W/kg and 250-350 W-yr/kg. Extrapolations of all types of past system technologies demonstrate that these goals seem reasonable with newer, more advanced technologies. It is only necessary to determine the level of technology advancement necessary to meet real mission needs.

MISSION CONSIDERATIONS

Reactor power systems have always been directed at next-generation, far-term missions. In the early 1960s, reactor-powered space station concepts were being studied (Figure 24). As power system concepts and missions evolved, the 1970s envisioned reactor-powered lunar and space stations (Figure 25). In the 1980s, the Galileo mission to explore Jupiter is being planned (Figure 26).

Today, the most important activity for the space nuclear power community is determining if any mission requires a reactor power system. As a starting point in the search for a mission, we can note the Soviet Union's use of reactor power for RORSAT. Surveillance is a potential mission with a potential near-term need. A possible technology for this mission is the synthetic aperture radar, which had its initial demonstration in the shuttle imaging radar (SIR-A) experiment. SIR-A was built by the Jet Propulsion Laboratory as part of the OSTA-1 (Office of Space and Terrestrial Applications)

experimental package (Figure 27). SIR-A operated at an altitude of 260 km (140 n. mi.). Interesting views from SIR-A include an oil slick near Venezuela (Figure 28), oil derricks near Abu Dhabi in the Persian Gulf (Figure 29), and freeways in the Los Angeles basin (Figure 30). SIR-A required a power of only 1 kW(e). It does not take much imagination to envision the capabilities of a near-term radar system having 20 or 30 kW(e) of power.

A radar surveillance system is only one potential mission. It does, however illustrate one truism--mission power needs will increase with time, and power system capabilities must follow those needs. As the power needs continue to increase, the necessity for reactor power systems will increase. Looking again at the mission "requirements space" (from Figure 7) with a time variable imposed, one can see a possible progression of power system capabilities over time (Figure 31). This time dependence is different for steady state and semicontinuous missions. In the late 1980s, there could be near-term missions, and only certain power system technologies may be capable of meeting near-term mission needs. In the early 1990s and mid-1990s, there could be midterm and far-term missions. Midterm and far-term reactor power systems will be available to meet midterm and far-term mission needs only if proper planning is performed now. This planning is the technology road map that fits time-phased mission needs to realistically achievable power systems.

TECHNOLOGY-MISSION OVERVIEW

The determination of such a technology road map was recommended to the Department of Energy by Rockwell International. The Energy Systems Group of Rockwell International is currently performing a technology-mission overview study to determine this road map for the Department of Energy. The study uses a "bottom-up" approach to assess power system capabilities and ties these capabilities to validated mission needs generated by the Department of Defense and the National Aeronautics and Space Administration (Figure 32).

An important feature of the study is technology assessment. In the "bottom-up" study approach, reactor power system technologies are first assessed at the subsystem level. Five subsystems are defined for this assessment (Figure 33). For each subsystem, current technologies are assesed, as are the future capabilities of improved technologies for midterm needs and advanced technologies for far-term needs. Of prime importance is the consideration of development time and development cost for each improved or advanced technology. When complete, the technology assessment provides time-compatible building blocks, (i.e., subsystems whose development can be achieved in the same time frame. These building blocks are then used to generate systems whose schedular availability can be time phased by near-term, midterm, and far-term initial operating capability (IOC).

At the same time, the study provides a review of near-term, midterm, and far-term power system capabilities required by the users --the civilian and the military users. The needs of the users are correlated with the schedular availability of the power systems. Power system requirements are iterated with the users' mission needs (Figure 34). Considerations of development cost and development schedule will then result in a preferred technology road map for reactor power systems (Figure 35). The technology road map will correlate cost to schedule and need. The generation and use of this technology road map will ensure that space nuclear power is no longer 20 years ahead of its time (Figure 36).

FIGURE 1 The Space Shuttle: A new era in U.S. space technology.

FIGURE 2 The Space Shuttle: A reusable launch vehicle.
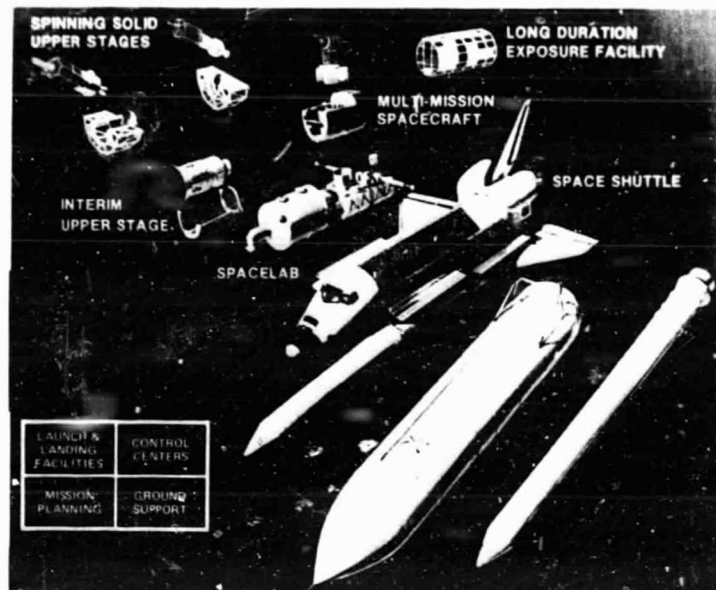
FIGURE 3   The Space Transportation System (STS).
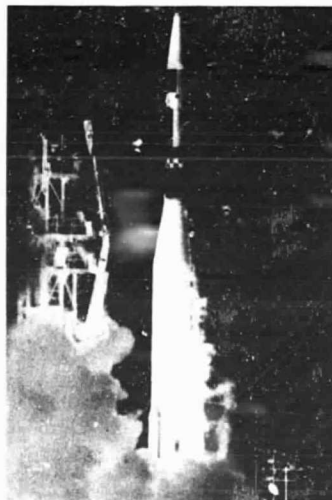
FIGURE 4   Launch of SNAP-10A, a single space reactor power system, on April 3, 1965.
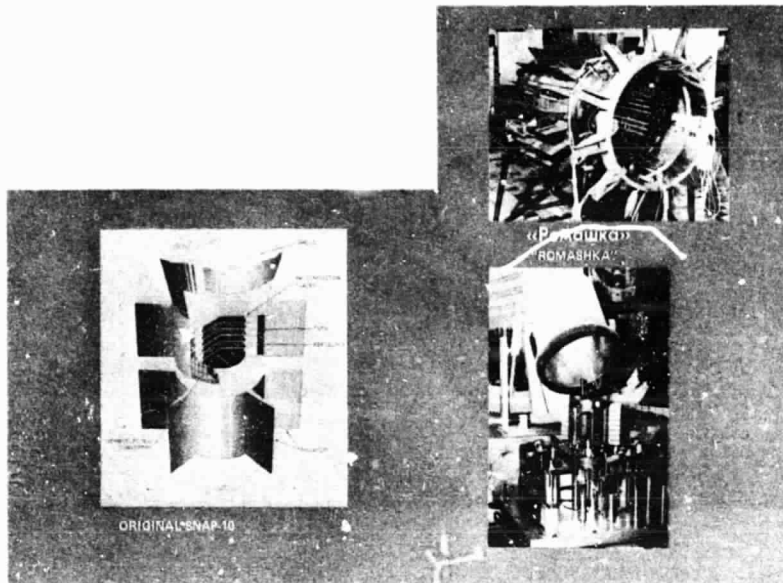
FIGURE 5 The early static SNAP-10 concept: model for USSR "Romashka" space reactor system.
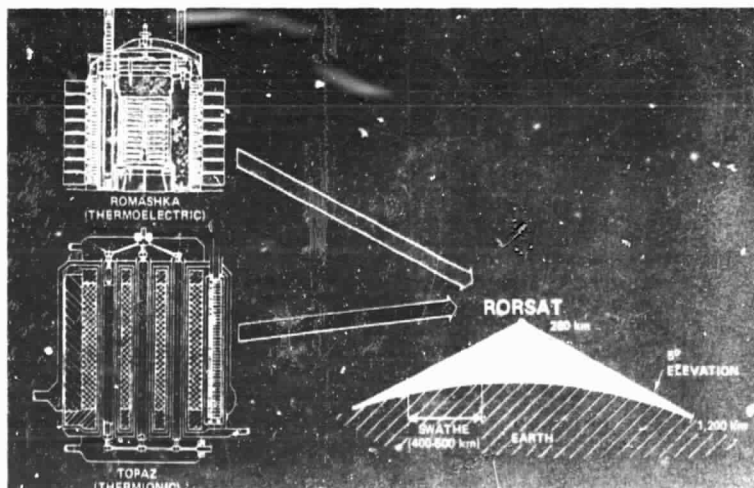
FIGURE 6 USSR RORSATs: using a reactor power system.
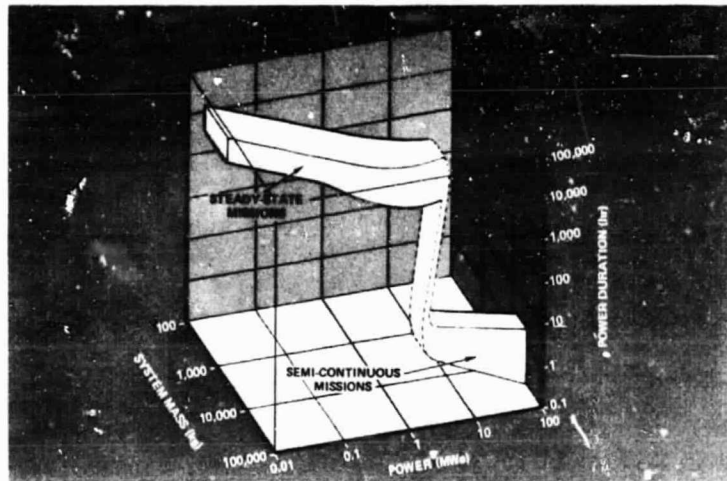
FIGURE 7   U.S. mission requirements.

FIGURE 8   SNAP reactor systems.
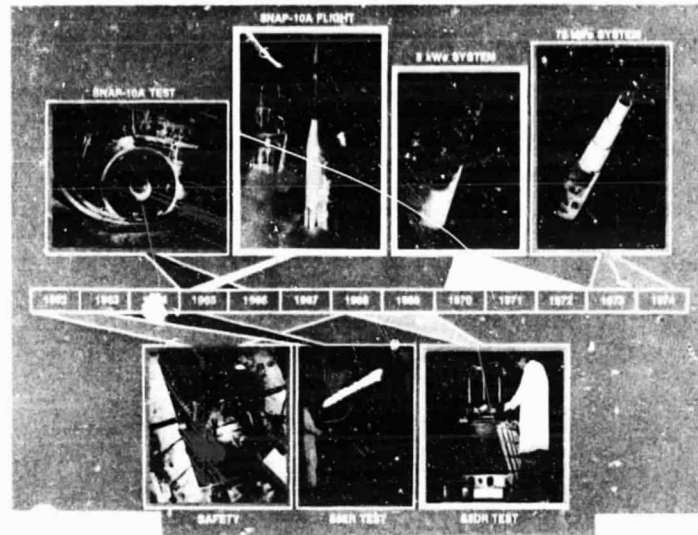
FIGURE 9   Major milestones in the SNAP program.
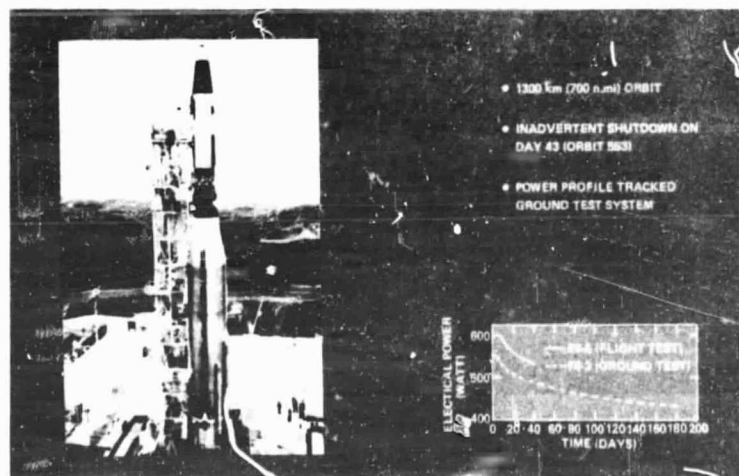
FIGURE 10   Launch of SNAP-10A Flight System 4 plus Agena spacecraft on April 3, 1965.
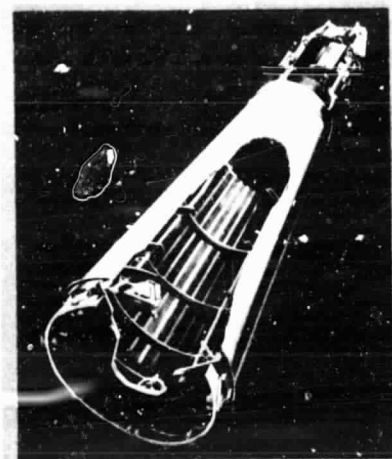
FIGURE 11   SNAP 10-A Flight System 4.

FIGURE 12   SNAP 10A Flight System 3.
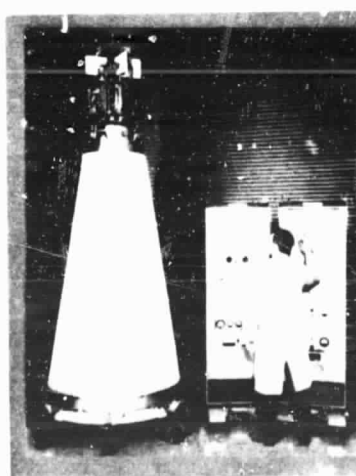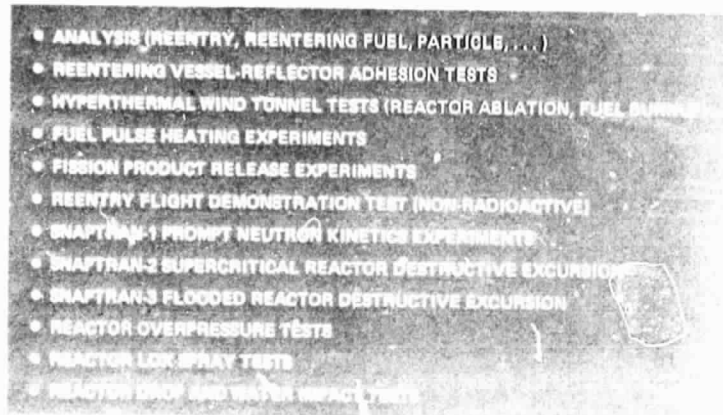
FIGURE 13   The safety program that supported the SNAPSHOT launch.
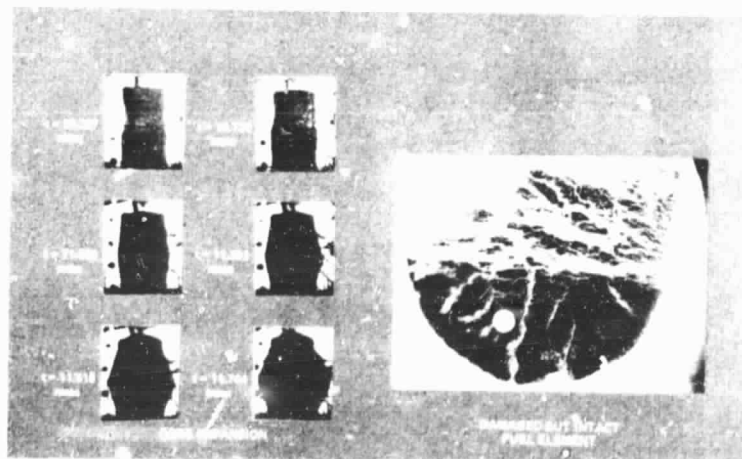
FIGURE 14   The SNAPTRAN-3 experiment.
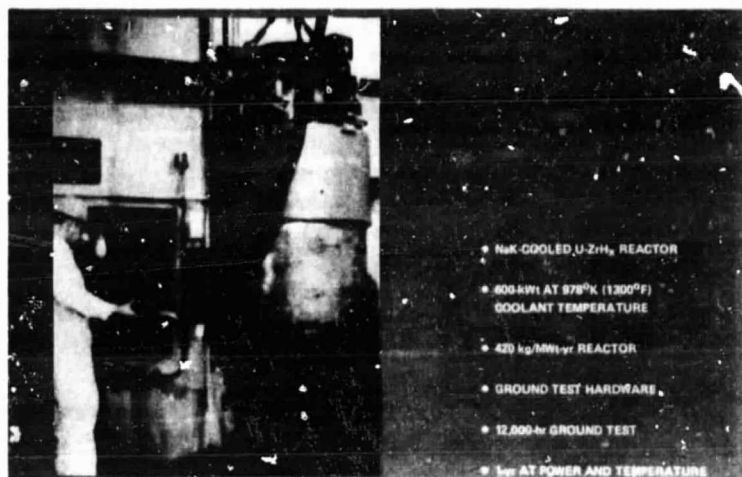
FIGURE 15   The S8ER: installation into test pit.
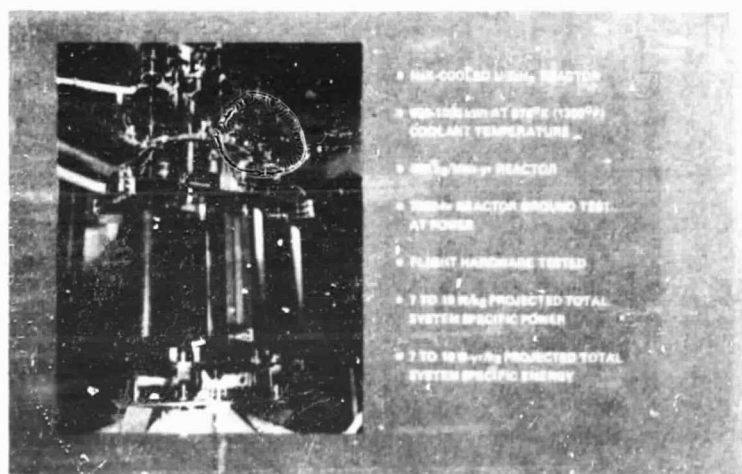
FIGURE 16   The S8DR ground test.

FIGURE 17 Follow-on SNAP reactors.
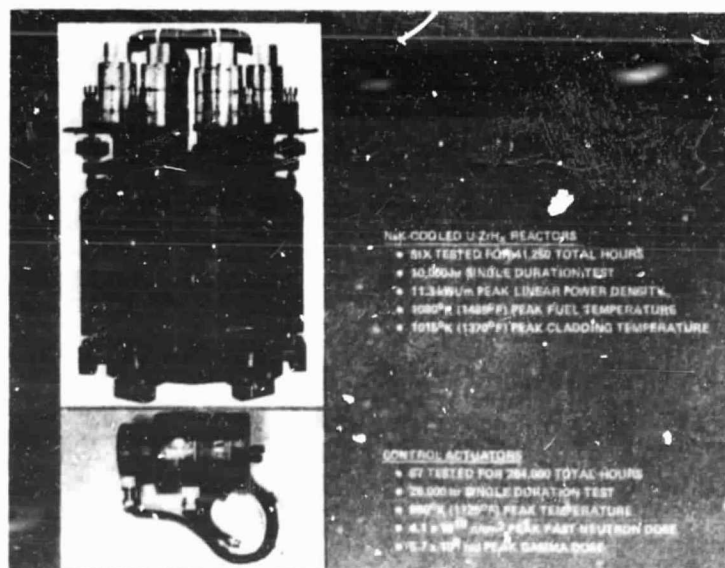
FIGURE 18 SNAP program data base: reactor subsystem.
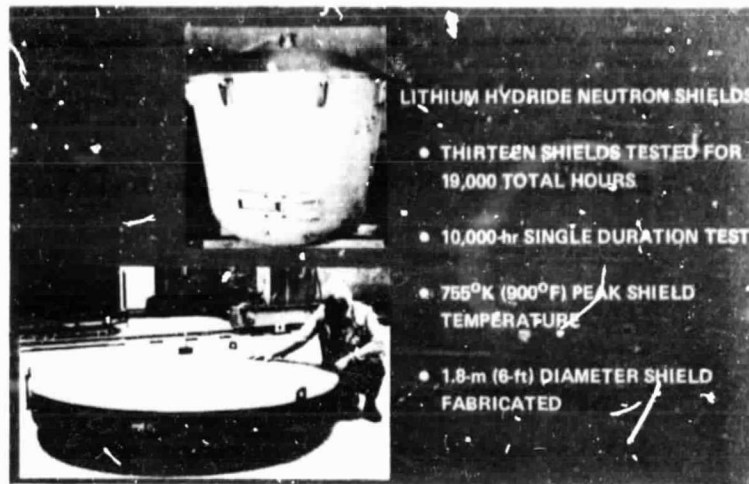
FIGURE 19 SNAP program data base: shielding subsystem.

FIGURE 20 SNAP program data base: primary heat transport subsystem.
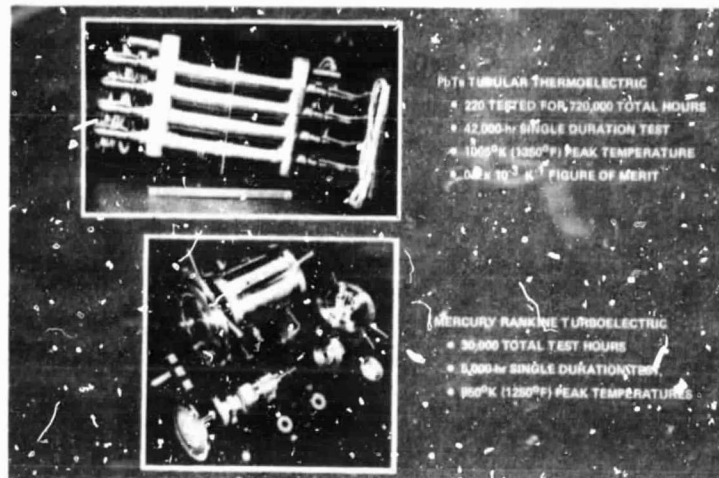
FIGURE 21 SNAP program data base: power conversion and processing subsystem.
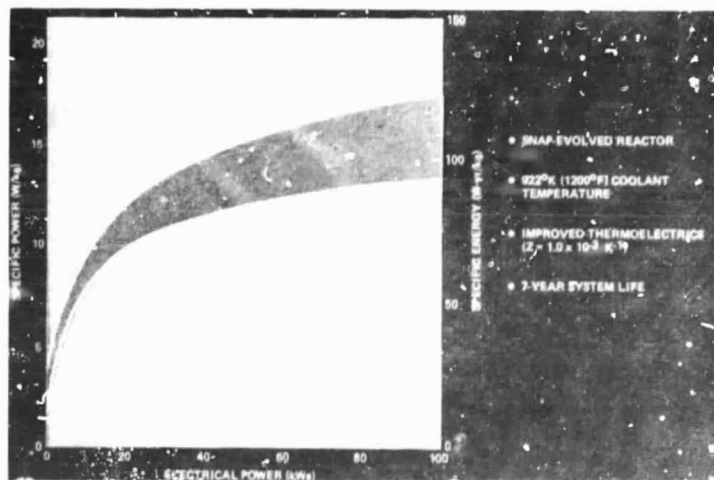
FIGURE 22 SNAP-evolved reactor concept, using some improved space power technologies.
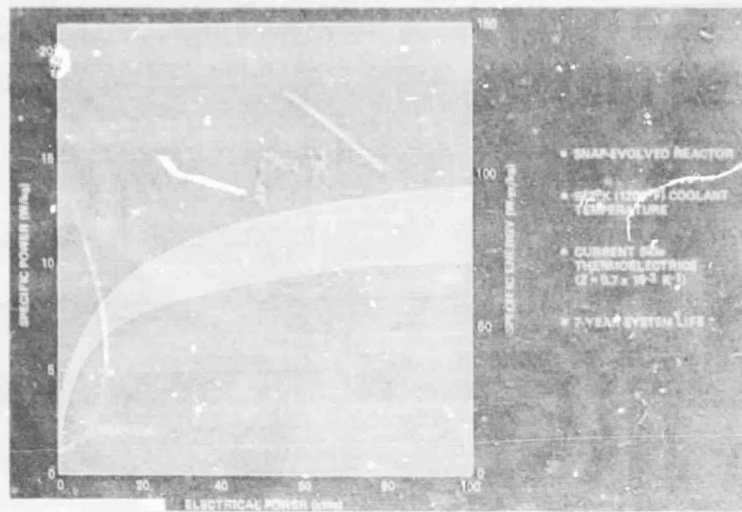
FIGURE 23 SNAP-evolved reactor concept, using improved power
conversion technology.
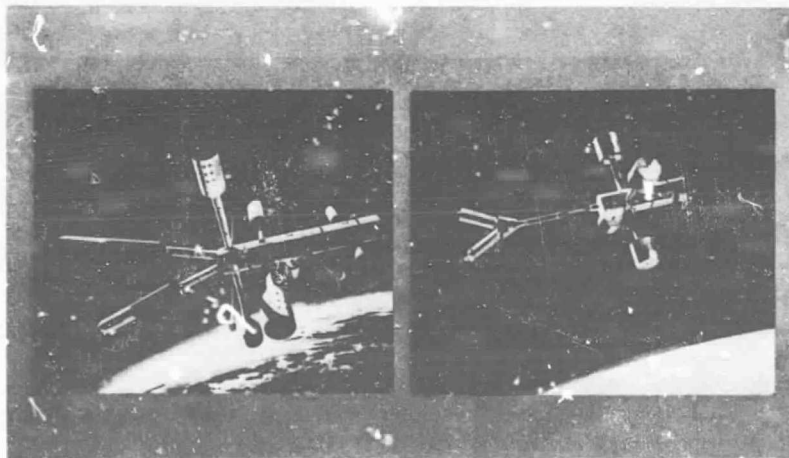
FIGURE 24 Early space station concepts.
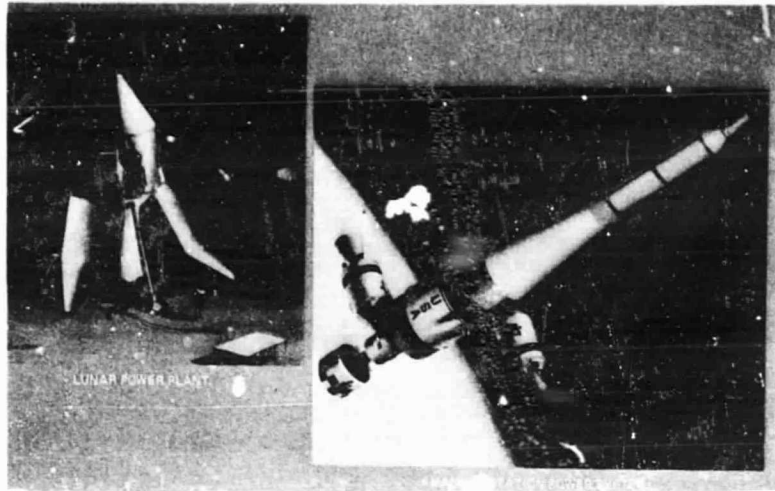
FIGURE 25   Lunar and space station reacto. power systems.
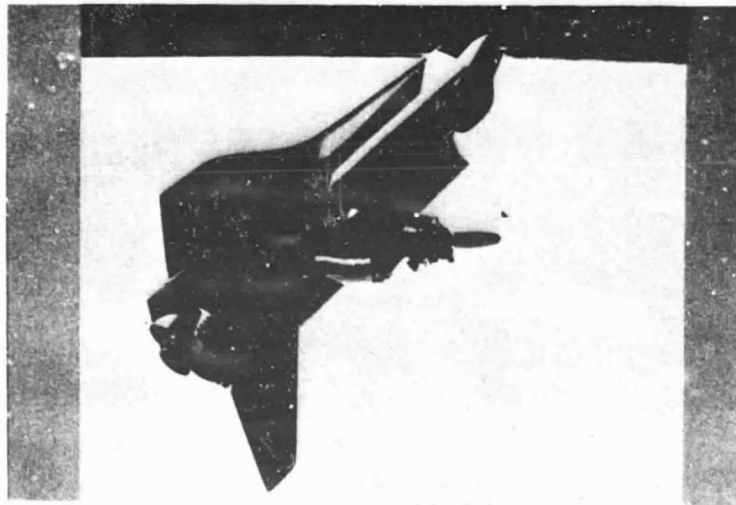
FIGURE 26   Future missions: the Galileo mission to explore Jupiter.
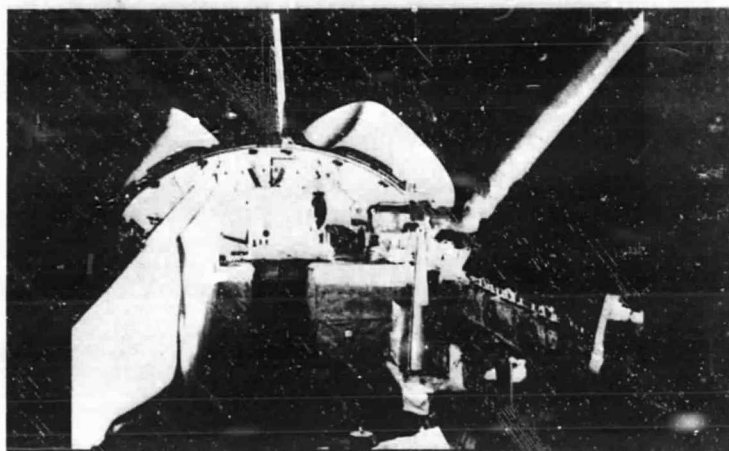
FIGURE 27   The OSTA-1 SIR-A, as demonstrated in STS-2.
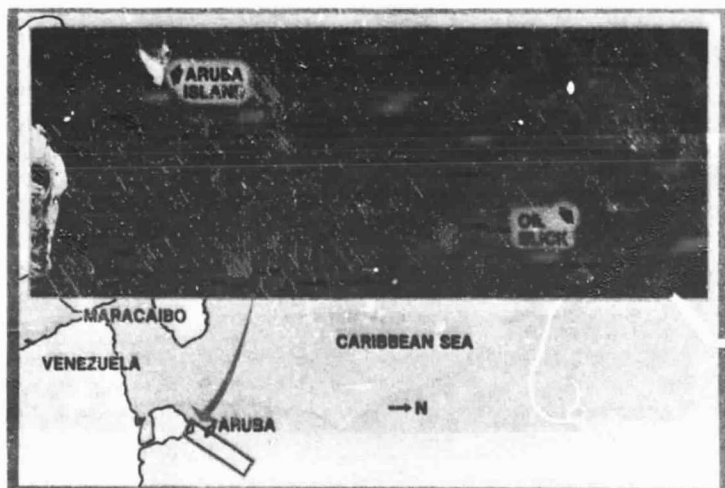
FIGURE 28   Oil slick off Venezuela spotted by SIR-A.

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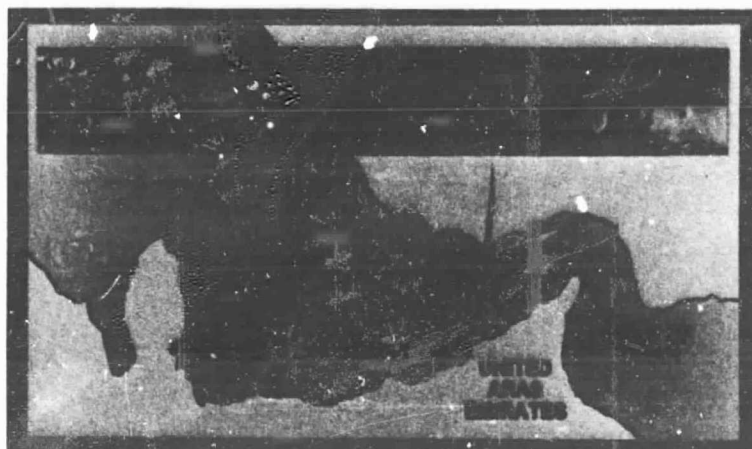
FIGURE 29 SIR-A strategic overlook of the Persian Gulf: oil derricks near Abu Dhabi.

FIGURE 30 SIR-A view of the Los Angeles basin, using only 1 (kW(e)).
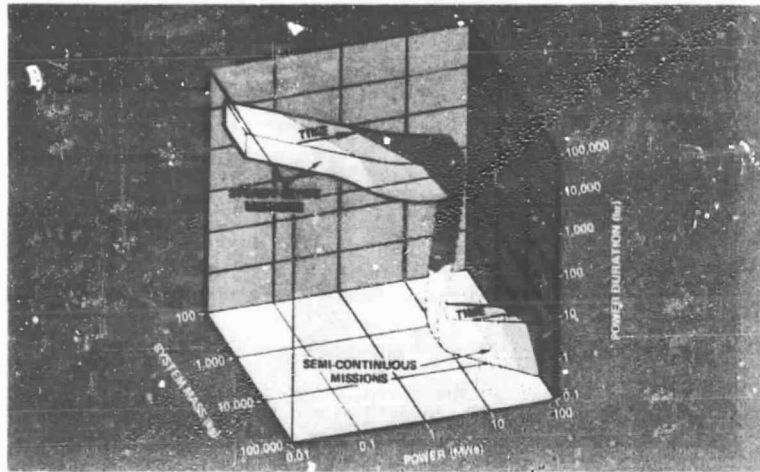
FIGURE 31 Possible progression of power system capabilities over time.
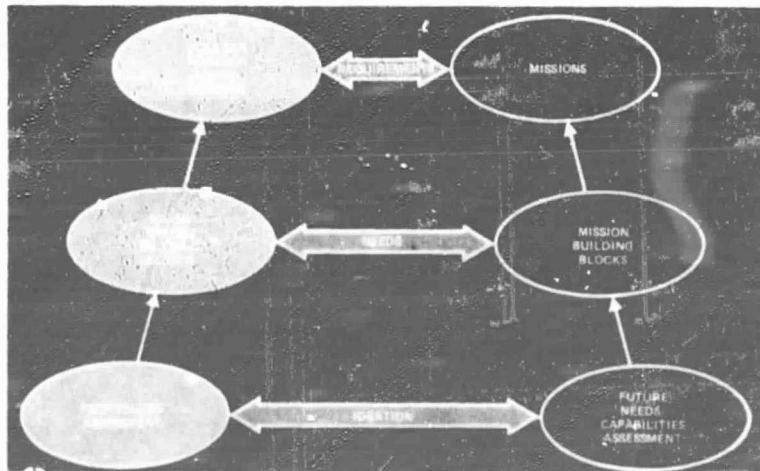
FIGURE 32 Assessment of reactor power system capabilities by use of
"bottom-up" analysis.
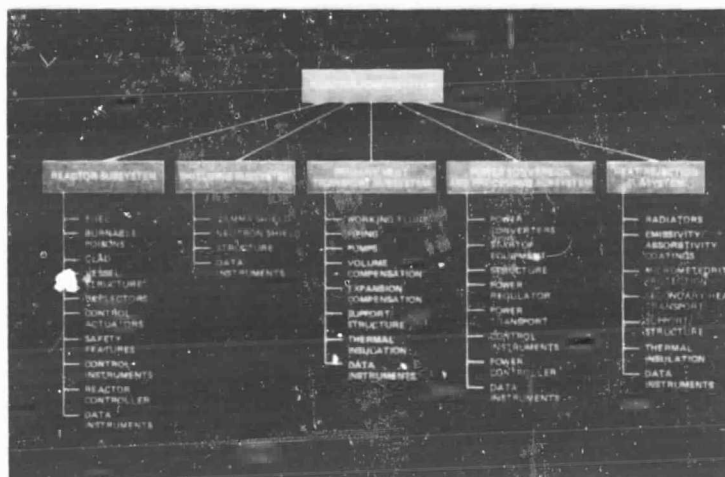
FIGURE 33 Assessment of reactor power system technologies, starting
at the subsystem level.
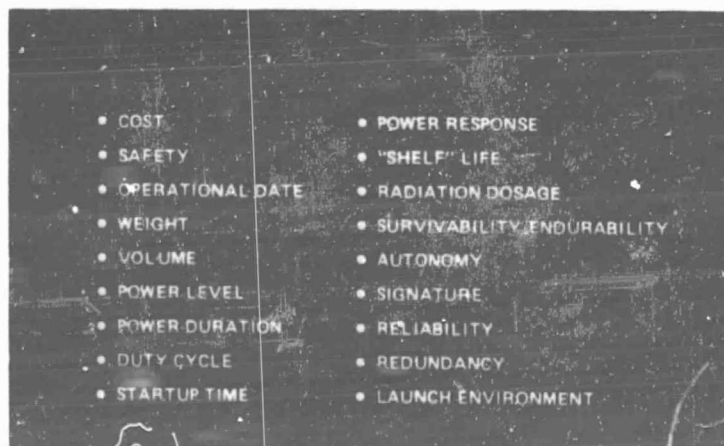
FIGURE 34 Iterating power system requirements repeatedly with users'
mission needs.

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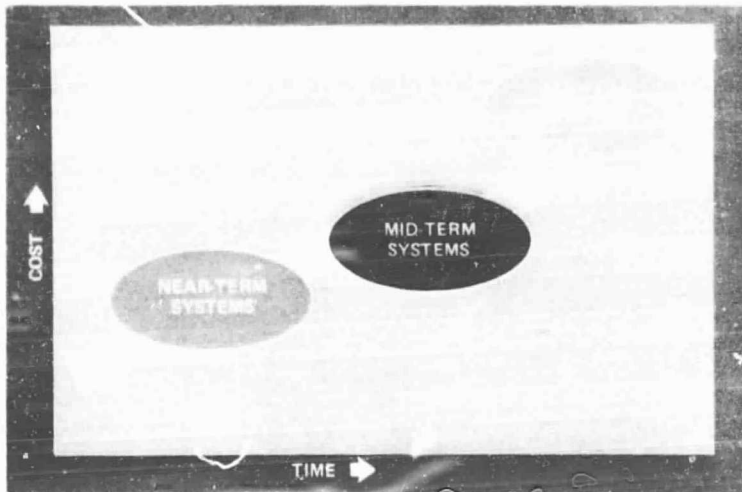
FIGURE 35 The preferred technology road map: a result of the
"bottom-up" analysis.
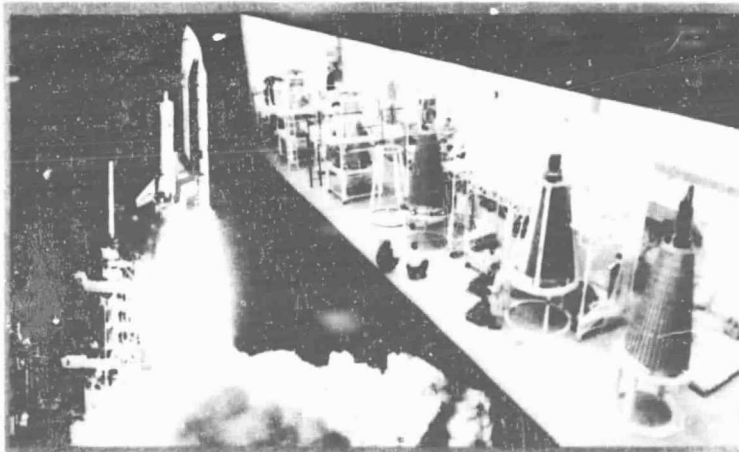
FIGURE 36 Space nuclear power: no longer 20 years ahead of its time.

SAFETY AND REGULATORY ISSUES
RAISED BY SPACE APPLICATIONS OF NUCLEAR REACTORS

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SPACE REACTOR SAFETY STRATEGY

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ABSTRACT

As needs develop for nuclear reactors in space, so does the need for a
comprehensive safety strategy. Although much of the strategy is
already in effect, important evolutionary extensions of current
practices and concerns are required for future implementation. This
paper proposes methods of achieving that goal, based on the two
fundamental objectives of (1) ensuring that reactor operations do not
expose the public, the mission personnel, or the biosphere to undue
risk and (2) demonstrating safety so convincingly that general
acceptance is assured. The strategy outlined addresses institutional
and organizational concerns, physical and engineering aspects, and
operational policy. The nature of the potential risks and the nature
of the events that embody those risks indicate that a continuity of
safety must be provided through the entire life cycle of the reactor
and across the many organizations involved in the different life cycle
phases. The physical and engineering practices are generally well
developed, but the design and implementation of a safety demonstration
program, of which the major aspects are to be understandable by a
rational lay public, will be challenging.

INTRODUCTION

In considering the safety of nuclear reactors for use in space
applications, we must address the relevant technical, biological, and
even social questions because the answers will play a major role in
determining the future use of reactors in space. Space reactor safety
has involved case-by-case considerations for more than two decades.

In the last few years, further attention has been given to the subject in the light of new reactor technology, new vehicles and missions, and new social and political realities; but a comprehensive, agreed-upon, and articulated strategy has not yet emerged. It is the purpose of this paper to propose a step in that direction.

In stating the U.S. Department of Energy (DOE) position for the United Nations Working Group on Nuclear Power in Space, Dobry (1979) summarized the status of that development:

> Basic U.S. safety philosophies relative to space reactors have not been developed or reexamined relative to new reactor design and launch vehicle capabilities, since no requirements have been forthcoming for 13 years. Should a requirement be established vital to the national interest, design safety philosophies would have to be developed; an estimated two-year period would be required to address overall safety objectives and criteria and a comprehensive and responsive program for the safety protection of the public and the environment. Past U.S. practice for all space nuclear systems has been established to specific requirements on a case-by-case mission basis.

A strategy must be developed to achieve objectives that are at least generally framed. Two objectives are proposed here. The first is generally recognized and can be formulated from statements by Dobry and from those in the "DOE Nuclear Safety Criteria and Specifications for Space Nuclear Reactors" (Department of Energy (DOE), 1982) as:

1. There shall be no undue risk to the public, to mission personnel, or to the environment, resulting from the use of space nuclear reactors.

The second objective is not generally stated, but I believe it is crucial to the long-term viability of a space reactor program. It is:

2. The safety of any space nuclear reactor must be so well demonstrated that general acceptance is assured.

The concerns of various public constituencies must be addressed if we are going to obtain long-term political support for space reactor programs.

In addition to these two objectives, which form the foundation for the strategy that is developed and presented here, it is also necessary to consider the physical environment that is important to space reactor safety, the major types of risks, and the events that can lead to those risks. A review of these factors provides focus and direction for the development of an appropriate strategy. Also, implementation is vital if a strategy is to be effective, and will be considered as the strategy is developed. In so doing, elements of the

strategy that already exist will become apparent, as will developments tnat are important if both objectives are to be met.

PRINCIPAL PHYSICAL CONSIDERATIONS

Four fundamental types of risks are involved in the use of space reactors:

1. There are radiological risks related to the inventory of radioactive materials, including fuel, fission products, and activated materials.

2. There are chemical risks related to the use or development of toxic materials in the reactor construction and operation or to the use of materials that, owing to tneir chemistry, increase the risk of fire or other chemical reactions.

3. There are mechanical risks that include tne impact of dense materials and tne deposition of energy in such forms as heat and pressure waves.

4. There are secondary or incremental risks that relate to the enhancement of otner risks owing to the presence of the reactor (e.g., the chemical nature of reactor materials could have adverse effects on the launch vehicle or its systems).

These various types of risks can occur singly or in combinations, and botn the single occurrences and tne combinations are functions of the specific events tnat lead to tneir actuality.

The nature of the events that lead to risk and the environment in wnicn tnose risks occur are related to the specific phase of the space mission. So far as tne reactor and its related systems are concerned, there are four major mission phases. The first combines manufacture, testing, transportation, and the assembly of tne reactor with the launch vehicle. The second consists of the various prelaunch activities anu tne suosequent launch to orbit (or other flight trajectory). The third phase is the space operation, the operative part ot the mission. Postoperational disposal is the last phase. Altnough a complete assessment of the possible accident events associated with each ot these phases requires careful analysis for specific cases, some general observations are possible.

The activities associated with manufacture, testing, transportation, and assembly are not unique to space reactors, with the exception of those related to launch vehicle integration. Well-estaolished practices govern manufacture and testing. Transportation is a normal activity involving well-known precautions of routing, containment, packaging, and subcritical component shipping. It is possible that reactor materials or systems could interact witn tne oalance of tne payload or with the launch venicle during integration activities to create a hazard, but this possibility

will normally be considered when establishing payload and launch system compatibility requirements.

During prelaunch activities, the primary concern relates to inadvertent criticality, with subsequent effects on mission personnel and the environment. The criticality could be caused by loss of control, such as the malfunction of control drum operating mechanisms, or by chemically or mechanically introduced reactivity, such as the accidental extraction of absorber plugs. Although these events are conceptually possible, they are relatively easy to deal with in design and through well-tested administrative procedures. The launch itself is more problematical, however, and it is necessary to consider both the normal, successful launch and the various types of launch failures.

During a normal launch, the reactor and its systems must survive without suffering any safety-related failure. The launch environment includes such adverse factors as vibration, temperature, shock, acceleration, and humidity. All of these factors are well known and are amenable to treatment in design and development.

Failed or abnormal launches include events such as launch pad explosions or fires, launch abort after lift-off, and failure to achieve orbit. In either the case of a launch pad explosion and fire or the case of launch abort after lift-off, the primary concerns relate to the disposition of a critical reactor or chemically active materials in uncontrolled locations. These launch accidents can create severe environments. For example, the proposed test conditions for launch pad explosion and fire qualification of the SP-100 reactor include a 2,000-psi overpressure, with an impulse of 5.4 psi-s, and a solid propellant fire subjecting the reactor to 2060°C for 10.5 min (Seabourn, 1982). Further, impact of the reactor on land or water is also possible. Under these conditions, criticality could be caused by deformation of the reactor, by damage factors such as the removal of absorber materials, or by immersion in water or some other fluid such as liquid hydrogen.

In the case of marginal failure to achieve orbit, the reactor may impact on land or water. Alternatively, it may break up in the atmosphere, resulting in various-sized fragments being distributed over a wide area and at various altitudes. Again, the major concerns relate to criticality and the distribution of radiologically and chemically active materials. Use of the Space Shuttle will reduce the possibility of these particular accident conditions, but they cannot be ignored.

Compared to the hazards attendant during launch, hazards during space operations appear minimal. At reactor deployment it is possible that various safety locks will require removal. The operation will be accomplished, directly or remotely, by Space Shuttle crew members. An inadvertent criticality at this point could therefore have safety implications for the crew. So far as postdeployment operations are concerned, there are only direct safety implications if the mission is manned or if it requires manned intervention such as servicing.

Generally, the types of hazards related to space operations appear amenable to system design and to operating procedures.

Postoperational disposal is a safety concern only if the reactor reenters the atmosphere. In this case, the achievement of reactor shutdown and the length of the delay between shutdown and reentry are both important because they influence the magnitude of the radioactive inventory at the time of reentry. The concerns associated with the reentry itself relate to reactor impact on land or water, related criticality in an uncontrolled location, and the possible release of radioactive or toxic materials. Similar concerns would also arise if reactor breakup should occur, resulting in a spectrum of fragment sizes, compositions, and deposition locations.

The nature of the risks associated with the use of space reactors and the nature of the events that embody those risks provide an insight into the structure and requirements of a safety strategy. First, safety must be a prime concern through the entire life cycle of the reactor, from conception, through design and development, through manufacture and operation, to final disposal. Decisions and actions taken as early as concept definition can affect safety as late as final disposal. Second, there must be a continuity of safety concern that spans the many institutions, agencies, and organizations that will inevitably be involved with the various reactor life-cycle phases. Third, although safety is of concern in all mission phases, particularly severe accident environments can occur during launch and reentry. Further, because these severe environments are coupled with launch and reentry, they are coupled with conditions and motions capable of placing hazards in a wide range of locations. Fourth, launch or reentry accidents would be "singular" or "dramatic" events. This fact may influence the accident frequency or the level of risk that the public will accept, as indicated by the concept of "risk aversion" (U.S. Atomic Energy Commission, 1975) and by the influence of irrational fears on public acceptance (Weinberg, 1982).

Finally, careful analysis and testing to a very high degree of survivability, and with impeccable credibility, are likely to be required to achieve appropriate safety and to demonstrate its achievement.

OUTLINE OF A SAFETY STRATEGY AND ITS IMPLEMENTATION

A successful strategy for the safety of nuclear reactors in space will have at least three major components. The first will address institutional and organizational concerns, the second will address physical and engineering aspects, and the third will address operational policy. The cohesive operation of all these components will be required if acceptable safety levels are to be achieved and demonstrated.

Institutional and Organizational Aspects

Safety is a prime concern in all phases of the reactor life cycle. Because many institutions, agencies, and organizations are involved throughout that life cycle, a single organizational entity must be responsible to provide focus and continuity to safety considerations. This entity must assure itself that the various national (and agreed-to international) radiological standards are met by a particular reactor in a given mission. These standards include, but are not limited to, those presented in DOE 5480.1A, Environmental Protection, Safety and Health Protection Programs for DOE Operations, and 10 CFR 50, Appendix A. General Design Criteria for Nuclear Power Plants, as well as those recommended and agreed to by the United Nations International Commission on Radiological Protection.

Safety cannot be produced simply by review and mandate; it must actually be achieved in operation. Consistent with the provision of safety continuity through all phases of the life cycle, the responsible organization must also concern itself with operational organization, procedures, and such factors as personnel selection and training.

This organization must ensure that realistic assessments are made of the risks associated with the use of a specific reactor in a given mission. These assessments involve appropriate reviews, analyses, and safety experiments. This organization must also ensure appropriate independence between those organizations involved in these activities and those responsible for program activities such as system development and mission accomplishment. This responsibility is important even early in the concept selection and design, because safety information may constitute critical design information, especially if the policy of inherent safety, presented later, is to be successful.

It is also important that the local organization satisfy itself that the general safety review and any critical analyses or tests are understandable by a rational lay public.

Finally, this entity must undertake the first and most detailed value judgment implicit in satisfying the goal that a space reactor mission is not accompanied by "undue" risk.

At present, the launch of a space reactor by the United States is approved on a case-by-case basis, with the authorization being given by the President. Questions of safety form a central part of the review that leads to launch approval or denial, and the organizational entity responsible for consideration of those questions is the Interagency Nuclear Safety Review Panel (INSRP).

The relationship between the panel, those organizations providing safety information, and the pertinent government agencies is shown in Figure 1 (Bennett, 198_,. The INSRP consists of representatives appointed by the Secretary of Defense (DOD), the Administrator of the National Aeronautics and Space Administration (NASA), and the Secretary of Energy. Other agencies such as the Nuclear Regulatory

Commission (NRC), Environmental Protection Agency (EPA), and National
Oceanic and Atmospheric Administration (NOAA) also participate.

The INSRP issues an independent safety evaluation report and a
recommendation for launch approval or denial. A recommendation for
approval indicates that the representatives of DOE, DOD, NASA, and the
other participating agencies believe that the launch and operation of
the space reactor is free of "undue risk."

The existence of the INSRP satisfies the major organizational
strategy requirements cited earlier. It provides a single focal
entity that spans the various organizations involved in the various
reactor life cycle phases. The panel requires and uses realistic risk
assessments and also considers the application of appropriate
radiological standards. Further, the panel has the ability to ensure
the consideration of safety early in the reactor development, at the
concept definition phase. With the representation of various
government agencies, it has the ability to ensure appropriate
independence in safety studies and can bring appropriate resources to
bear on those studies. It also undertakes the value judgement
associated with launch approval. In the future, with the use of
larger reactors and with the advent of more complex total missions, it
may be necessary for the activities and concerns of the panel to
evolve in the direction of operational considerations even more than
has occurred in the past. Further, the panel is the logical
organization to consider acceptance of space reactors by the various
public constituencies. In general, the INSRP provides for the
organizational considerations of the strategy, but full implementation
in the future will probably require some evolution of the activities
and concerns.

Physical and Engineering Aspects

The physical concept and design of a reactor may produce "inherent
safety," because it is physically impossible for certain events to
occur. Other design aspects may enhance safety by reducing the
probability or severity of hazards in a passive or active manner.
Considerations of predictability and reliability should influence
design in the direction of inherent safety and in the direction of
passive systems over active ones. Therefore the overall safety
strategy requires that inherent safety be a prime goal in concept
definition, system design, and system development. Deviations from
inherent safety should proceed from passive to active systems.

Ideally, implementing the inherent safety strategy lies in
designing a reactor that is intrinsically subcritical under all
accident conditions. The reactor should remain subcritical in the
event of deformation or damage resulting from mechanical forces such
as explosive overpressure and impact or from heat due to a launch
accident or reentry. Intrinsic subcriticality is also a design
objective with regard to immersion and flooding of the reactor.

Ideally, the reactor should remain subcritical even if this immersion or flooding is coupled with damage to the reactor or its controls.

The general idea of intrinsic subcriticality has been discussed by Buden and Bennett (1982) and Bennett et al. (1982). The influence of requiring subcritical immersion on the core envelope dimensions of a uranium-zirconium hydride reactor with a beryllium reflector was investigated by the Atomic Industrial Forum (1968). The results indicated that high core length-to-diameter ratios were required to provide subcriticality in water with a beryllium reflector of reasonable thickness, as demonstrated in Figure 2. Although other reactor types may not produce such geometrically restrictive requirements, there is certainly an indication that inherent safety may not always be practically achievable, particularly with regard to all the events related to launch and reentry.

Passive systems can be used to enhance reactor subcriticality duri   launch, with removal of such systems after launch. Use of the Space Shuttle enhances this capability. As an example, an integral bottom core disk and central plug of boron carbide will be used in the SP-100 reactor for that purpose (Seabourn, 1982). Further examples are passive locking systems that immobilize rotating control drums until after orbit is achieved.

Passive systems have also been proposed and used to minimize risks associated with the reentry of a reactor after some period of operation (i.e., with some remaining radioactive inventory). Design of the reactor so that reentry heating causes breakup, with the ultimate impact of nonrespirable particles, has been proposed as an effective way of minimizing risk (Bartram and Pyatt, 1979), and, in fact, DOE (1982) requirements call for the use of passive systems to achieve such a breakup.

Although such a breakup has the potential to minimize risk, it may not meet the second objective of the safety strategy, that of obtaining general acceptance, especially if numerous reentries are planned. It is possible that this second objective may direct efforts toward total containment of radioactive materials within an intact boundary or boundaries. However, Bartram and Pyatt point out that uncontrolled, intact reentry could present safeguards problems and that intact reentry of a large enough reactor on land could result in high radiation doses to people in the immediate vicinity (Bartram and Pyatt, 1979). Evidently, the questions related to reentry are difficult and bear further thought.

Active systems for safety steps such as core dispersal have also been proposed and used but, according to the safety strategy, would constitute a last resort. Explosives have been considered for core dispersal, and the Soviet Union employs chemical dissolution for reentry breakup of its space reactors (Buden and Bennett, 1982). The dissolution may be semipassive or active, depending on the initiation mechanism.

Active systems have also been proposed and used for termination of reactor operation. For example, reflector control drums can be

actively disassembled to ensure reactor shutdown. When such systems are employed, they should be fail-safe in nature.

The hierarchy of safety using physical limitation, passive systems, and active systems is evidently well used, but continuous organizational discipline is required to ensure that all the possibilities at each stage have been exhausted before the next stage is allowed.

The second element of the strategy related to physical and engineering aspects is that the success of reactor safety features or the achievement of safety objectives should be as independent of mission as is possible. Mission factors (such as orbital characteristics) should not be used to achieve basic safety requirements. This strategy element is based on the proposition that failure to achieve a given mission flight profile should not cause a fundamental degradation of safety. However, this general approach does not preclude active attempts to enhance safety through mission design and control.

The most important example of safety enhancement by this means is through the selection of long-lived orbits to allow postoperation decay of the radioactive inventory to minimize possible release to the biosphere. The anticipated decay of fission product inventory following operation of the SP-100 reactor is illustrated i Figure 3 (Palmer, 1982), which demonstrates that postoperational orbital lives of about 300 years or more will reduce the radioactive inventory by 4 orders of magnitude. This result is consistent with Bartram and Pyatt's contention that long postoperational orbit lives are advantageous but that orbit cycle extensions significantly beyond 300 years do not reduce risks substantially (Bartram and Pyatt, 1979). Using a very conservative approach, the SNAP-10A reactor was placed in a 4,000-year orbit before it was started, to ensure decay to very low levels (Bennett, 1981). The geosynchronous orbit, which is of interest in a number of mission applications, offers an essentially indefinite orbital life.

Orbit life considerations can be complicated by postlaunch boosting. Boost from a lower orbit to a higher orbit can occur early in the mission or may be used as a disposal technique. For example, a reactor could be placed in a low earth orbit by the Space Shuttle and chemically boosted to a higher orbit. The reactor may then be used to provide power for nuclear electric propulsion to even higher orbits. Buden et al. (1980) indicate that if nuclear electric propulsion were initiated at an orbital altitude of 850 km, there would be very little radioactive inventory in the event of reentry. Figure 4 (Buden ec al., 1980) illustrates that if nuclear electric propulsion were initiated at orbital altitudes of 450 or 620 km, the radioactive inventory during reentry would first increase owing to operating time spent in low orbits, but that as the operating time and orbital altitudes increased, the resultant activity at reentry would diminish. Obviously, these considerations could become even more

complex if the reactor were started and multiple orbit changes were later required.

Boost from low orbit to high orbit has also been used for postoperational disposal. The current DOE (1982) space reactor safety criteria require an orbit boost system for the SP-100 reactor if it is used in low orbit, with the specification that the booster be able to provide orbits with at least a 300-year life. The Soviet Union regularly employs postoperational boost for disposal of its ROMASHKA reactors. The boost is typically from an orbit between 250 and 280 km to a 900-km orbit following their operational lifetime. The disposal orbit provides lifetimes of 100-1,000 years, depending on ballistic coefficients.

Clearly, mission flight path characteristics can be used to enhance safety effectively. Nevertheless, the reactor system should be fundamentally safe regardless of flight path.

The last element of the strategy related to physical and engineering aspects is that the risks associated with the use of a given reactor in a given mission should be realistically assessed using probabilistic safety analyses and related or supporting analyses and tests. For important focal aspects such as source term limitation, safety performance should be demonstrated by appropriate component tests and by appropriate full systems tests. This approach may extend to important safety systems that are not directly part of the reactor, such as disposal boosters.

The strategy element requiring the assessment of risks through analyses is already implemented by the various responsible agencies, and the results of such analyses are required input to INSRP documents. The methods employed are provided by the Overall Safety Manual (NUS Corporation, 1981). Basically, the analyses consist of three steps:

1. Determination of detailed mission events having the potential for causing exposure and the determination of the related probabilities or frequencies of those events.
2. Determination of the consequences in terms of numbers of persons exposed to various levels of radiation and the effects of that radiation.
3. Evaluation of the nuclear system on the basis of risk (a combination of probability or frequency with consequence).

These analyses are not limited to single events or to low probability events with high consequences. The more probable sequences of events are examined even if they result in nominally lower consequences.

So far as reactor testing is concerned, various tests have been used or proposed to evaluate source terms and reactor system behavior in various adverse environments. In the case of the SNAPTRAN-2/10A reactor, tests were conducted on mechanical reactivity insertion by rapid control drum rotation (Neal, 1965) and for the case of water

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immersion (Cordes et al., 1965). Various component tests are
envisioned for the SP-100 reactor related to such factors as explosive
overpressure and fire, among others (Seabourn, 1982).

In general, the requirement for analysis appears to be largely in
hand, but with the limited reactor development that has occurred thus
far, there is little experience with the implementation of a
demonstration test program involving both components and full systems.
We recognize that the costs associated with implementing this approach
will likely force us in the direction of employing a relatively small
number of standardized components and designs, resulting in a modular
approach to meeting operational requirements (such as variations in
power requirements).

Operational Policy Aspects

In this context, operational policy simply refers to operational
choices in the approach to the use of reactors in space. The first
step requires that a policy of phased introduction be adopted.
Implementation of this policy lies in selecting missions with high or
geosynchronous orbits for the early use of space reactors. Missions
using lower orbits should be undertaken only after experience has been
gained with operation in the high orbits and after a level of
acceptance has been achieved.

The second step requires that operational procedures and methods be
developed to minimize risk by such actions as minimizing the
radioactive inventory at any given mission stage. The elimination of
prelaunch operation of the reactor, as in the case of the SNAP-10A
launch, is an excellent example of implementation.

Strategy Implementation

The institutional and organizational requirements of the strategy are
largely addressed by the existence of the INSRP. This body provides
for safety continuity across many organizations, has the potential to
ensure careful and credible safety analyses and tests, and undertakes
the primary value judgment to ensure that there is no undue risk
associated with the launch of a reactor on a given mission. In the
future, the concerns of the panel may evolve in the direction of
operational considerations and in the direction of ensuring that the
objective of safety demonstration to achieve general acceptance is
reached.

A hierarchy of techniques using physical limitations, passive
systems, and active systems is in effect and must continue to be
employed to maximize safety. Mission flight path characteristics can
be employed to enhance safety. However, both approaches will require
constant organizational discipline. In the first case, we must ensure

that all safety possibilities at a higher level are exhausted before
resorting to the easier paths of the next lower level. In the second
case, we must ensure that mission flight path considerations are not
used to justify relaxation of other fundamental safety approaches.

As far as safety analyses and tests are concerned, safety analyses
methods are known and regularly applied, although modeling
improvements will no doubt be made. However, significant additional
effort will be required, regarding a full reactor safety demonstration
test program, if safety is to be amply demonstrated. Owing to the
limited reactor development undertaken to date, little experience is
available for the design and implementation of such a test program.

The proposed policy of phased introduction of reactors in space is
considered to be important, and its ramifications should probably be
investigated further under the auspices of the Office of Science and
Technology Policy and the National Security Council, because these
agencies formally request presidential authorization for a given
reactor launch.

The basic policy of limiting risks in operation is evident in the
elimination of prelaunch reactor operation, but the concept requires
formal development for all mission phases and also requires
verification in operation. Once again, the evolution of INSRP
concerns in the operational direction is important to this strategy
element.

CONCLUSIONS

Fundamentally, reactors can be made capable of safe operation in space
applications. However, as the number of such applications increases,
the probability of encountering accident conditions that challenge
safety factors also increases. It is unlikely that a suitably low
level of risk will be reliably maintained in the absence of a
complete, agreed-upon, articulated, and implemented strategy to
achieve it.

An outline of a strategy has been presented, founded on the dual
objectives of ensuring that reactor operations do not place the
public, the mission personnel, or the environment in undue risk, and
of demonstrating safety so as to achieve general acceptance of the use
of reactors in space. The nature of the potential risks and the
nature of the events that embody those risks indicate that a
continuity of safety must be provided throughout the entire life cycle
of the reactor and across the many organizations involved in the
different life cycle phases. Further, severe accident environments
can be produced during launch and reentry, with the possibility of
placing hazards in a wide range of uncontrolled locations.
Consequently, very careful analysis and testing, to an extremely high
degree of survivability and with impeccable credibility, are likely to
be required to achieve appropriate safety and to demonstrate its
achievement. This strategy outline is aimed at achieving the major

objectives while recognizing the implications of the demanding physical environment.

In form, the strategy outline addresses institutional and organizational concerns, physical and engineering aspects, and operational policy. Much of the strategy is already in effect, but some important evolutionary extensions of current practices and concerns are probably required for full implementation in the future. The INSRP addresses the institutional and organizational aspects, but with the advent of missions requiring extended reactor operation and complex flight path adjustments, it will be necessary for it to address operational aspects of safety very carefully. The physical and engineering practices are generally well developed, but the design and implementation of a safety demonstration program, of which the major aspects are to be understandable by a rational lay public, will be challenging. The policies that govern the overall operational approach to the placement and use of reactors in space also require specific development.

Although a strategy outline is presented, it cannot be considered definitive. Dobry's assessment that a substantial, 2-year effort is required to address safety objectives and criteria along with a comprehensive safety program probably places this paper in a suitable perspective. Rather, the contribution lies in the thesis that a well-developed strategy is needed, that such a strategy must consider a variety of factors, and that to be successful it must be agreed to and articulated by the various institutions and agencies involved with space reactors. The early development of a complete safety strategy will probably prove crucial to the long-term viability of a reactors-in-space program and to whether the use of such reactors becomes commonplace in modern life.

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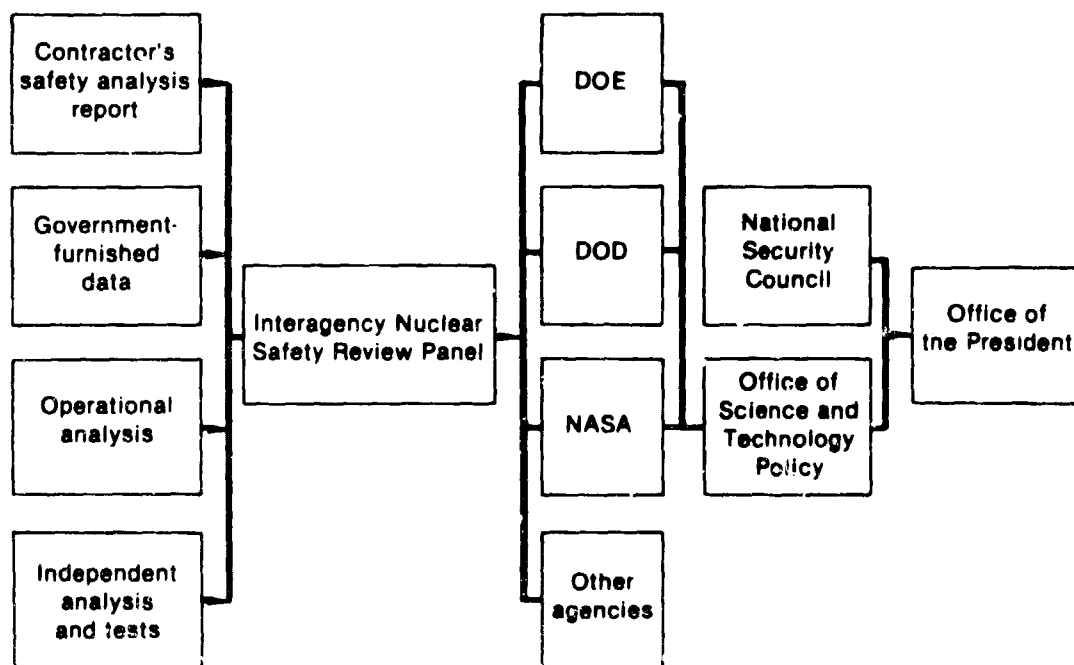
FIGURE 1   The INSRP and its organizational environment.
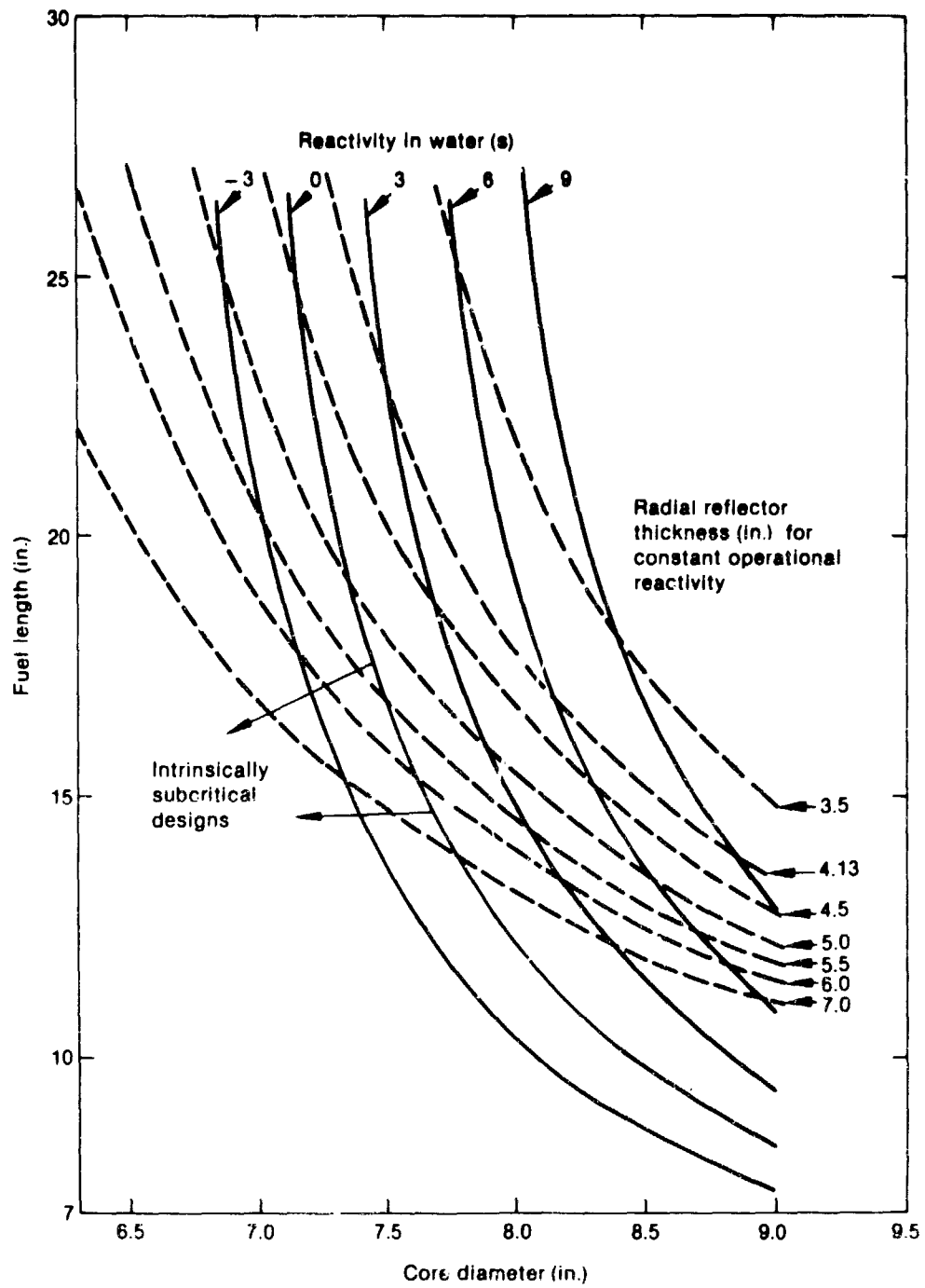
FIGURE 2 The effect of subcriticality requirement on core envelope geometry.

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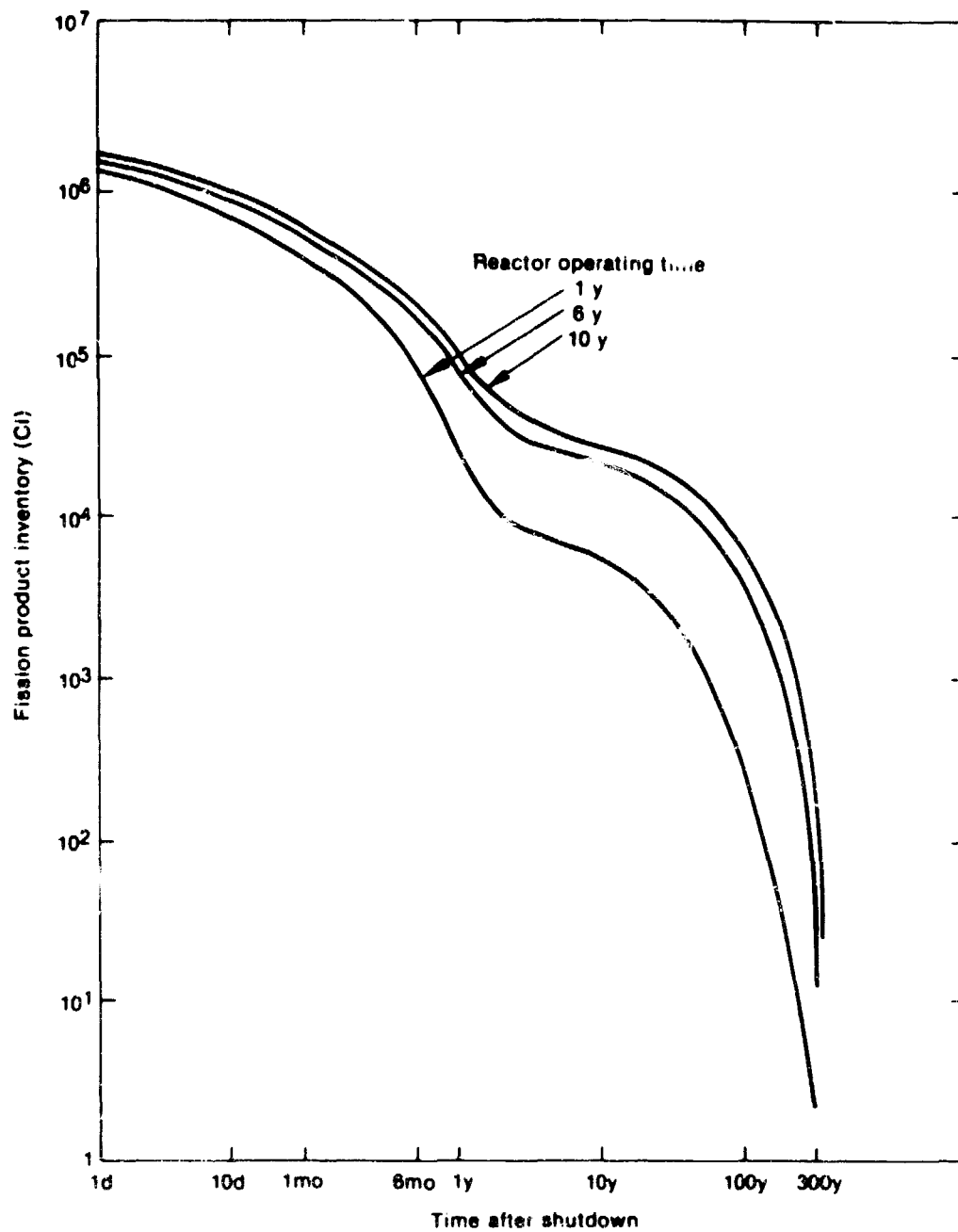
FIGURE 3 The reduction in radioactive inventory as a function of
postoperational orbit life for the SP-100 reactor.
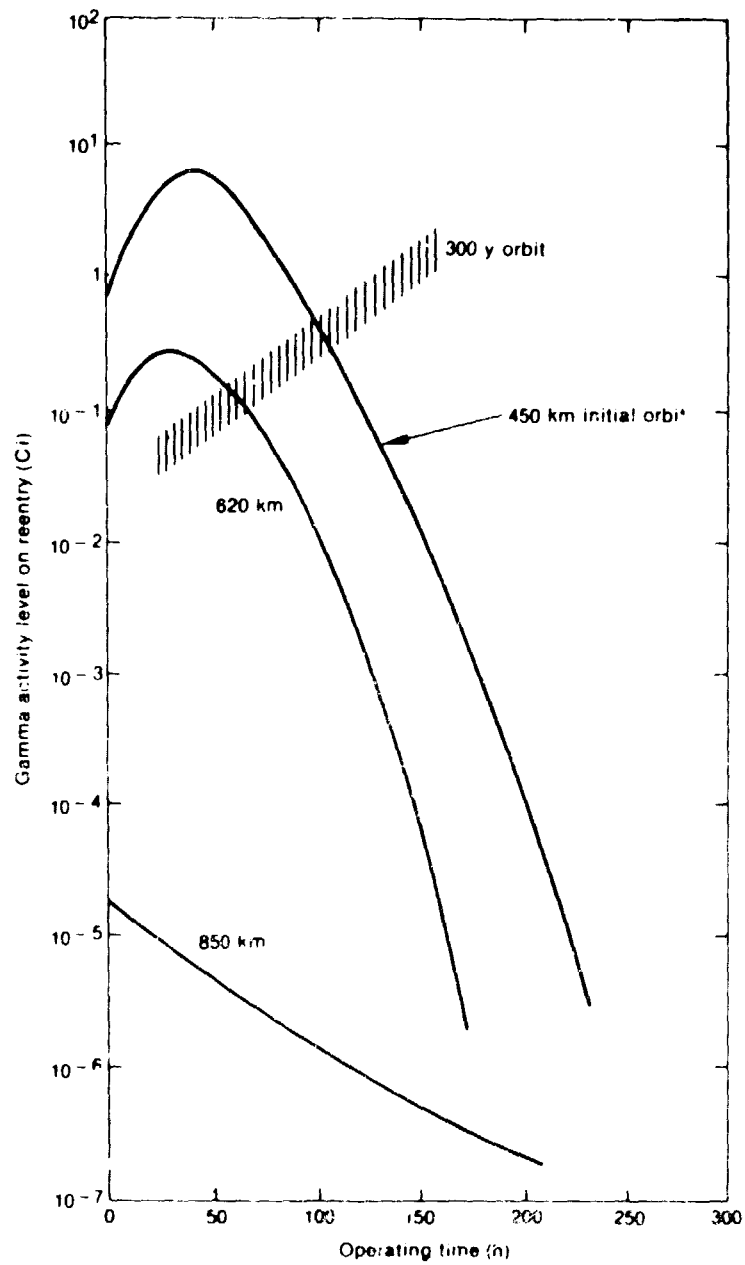
FIGURE 4 Reentry activity level.

FUTURE REGULATION OF SPACE NUCLEAR POWER SYSTEMS

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None of the 23 nuclear power systems used thus far to supply
electricity or heat for space missions has been subject to licensing.
The basis for exemption was obvious for the first five nuclear-powered
missions, which involved Department of Defense (DOD) navigational
satellites, and for two subsequent DOD missions. The Atomic Energy
Act exempts DOD from licensing when its acquisition or use of nuclear
materials or facilities is for military purposes.

Once the National Aeronautics and Space Administration (NASA)
entered the picture as owner and user of the space vehicles, the
situation became less clear-cut. The Atomic Energy Act provides no
exemption from licensing for government agencies other than DOD and
the Department of Energy (DOE).

Two reasons were nevertheless advanced to justify continuing the
exemptions. The first was that the development and production of
space power devices by DOE and its predecessors were in fulfillment of
the research and development responsibilities assigned to these
agencies in the Atomic Energy Act. One might conceivably question
this basis for exemption on the grounds that certain of the numbered
SNAP (Systems for Nuclear Auxiliary Power) systems were used several
times. SNAP-27, for example, was employed on six missions; SNAP-19 on
three. Did this repetitive use constitute an operational phase
distinct from research and development? The prevailing view appears
to be that it did not. Essential to this finding has been DOE's
avowal that in the development of succeeding SNAP devices, even those
that bore the same identifying number, there was a continuing effort
to improve performance and safety characteristics.

One can, however, envision a time when the technology will have
become so mature and so widely used that there may be repetitive
production by DOE of essentially identical models, or changes that
represent little more than production refinements not significantly
advancing the state of the art. Under such circumstances, it could be
argued that DOE would no longer be engaged in research and development
on these devices and that this basis for a licensing exemption for
users other than DOD or DOE itself would no longer be valid.

The second basis offered for claiming an exemption from licensing for the nuclear power devices on NASA missions has rested on the fact that by special agreement between the agencies, DOE retains title to the devices and their nuclear materials throughout the project. Further, since DOE personnel participate to the moment of launch, possession is also attributed to DOE. Consequently, the contention has been that the exemption provided in the Atomic Energy Act for DOE's own activities applies and that the projects are subject to DOE's internal health and safety controls rather than to the regulations of the Nuclear Regulatory Commission (NRC).

It might be asked why, in view of this basis for exemption, DOE has subjected its own Clinch River Breeder Reactor (CRBR) project to licensing procedures. The answer appears to be that this was not due initially to any legal requirement but to a deliberate government decision, one of whose purposes was to establish that a breeder reactor could be licensed in the United States. Subsequently, the full licensing apparatus of the NRC was applied to the CRBR.

Some disagreement has been reported within the government as to the validity of the second basis for exempting the SNAP devices in NASA missions based on DOE ownership and possession. According to one view, the fact that NASA assumes opperational control of the nuclear power systems at the moment of launch and retains it thereafter during all the period of risk has the effect of giving NASA possession under the terms of the Atomic Energy Act and would make the systems subject to licensing were it not for the research and development consideration.

I mention this legal controversy only because it may have implications for the future. It does not appear to have current significance. The next launches involving nuclear power devices are not scheduled until 1986, and the systems now planned appear to involve advances that would clearly qualify them for exemption under the research and development provisions of the Atomic Energy Act.

When we speak of licensing, we do not infer that its absence in the space nuclear power program to date implies any lack of appropriate safety requirements or of adequate analysis, review, and approval procedures. On the contrary, an effective interagency review process has existed from the outset. Since its formulation under presidential directive, the mechanism has been centered in the Interagency Nuclear Safety Review Panel (INSRP), with members from DOE, DOD, and NASA.

As a further protection of the public safety, the presidential directive also mandates the participation of NRC. There appears to have been a spirited controversy within NRC during the late 1970s as to how the commission should participate. One view was that the circumstances required NRC to undertake an independent review of the safety aspects of each proposed mission, even to the point of establishing an organizational unit for that purpose. An opposing view was that it would be sufficient for NRC to participate in the interagency review process as an observer, contributing to the process in its areas of expertise but without adopting an agency position

except in cases where it felt that a vital issue or concern was being mishandled in a serious way. The fact that this second view prevailed can be interpreted as a vote of confidence in the interagency review process by the NRC.

It should be added that while it is not required by the presidential directive to do so, the interagency panel has also enlisted the participation as observers of the Environmental Protection Agency and the National Oceanic and Atmospheric Administration.

The space nuclear power health and safety regime that has developed thus far has been successful. One indication of this success would seem to be the fact that although three U.S. space missions with nuclear power devices were aborted, there has been no evidence of any resulting human injury.

What of the future? There is a likelihood that future spacecraft will require larger amounts of power and hence larger fuel inventories. The hazards involved and the burden on the health and safety regime can be expected to increase proportionately. Nevertheless, if future space use of nuclear power sources continues to be very limited and noncommercial in character, it would seem reasonable to continue the type of review that has existed, centered in the interagency panel. On the other hand, if there is to be a greatly expanded program, not primarily experimental in character, including commercial enterprises and involving the launching into space of a singnificant amount of nuclear material, then we would have a different situation that could call for some changes.

The commercial ventures and perhaps some other uses may need to be licensed. In such a more active future for space nuclear power, some buttressing might particularly be needed of the system's truth-seeking capability with respect to the nature and the degree of risk. Appraising risks is an endeavor in which our society has not particularly distinguished itself. Witness the uncertainties that have been experienced in trying to reach assured judgments as to the degree of hazard involved in the use of pesticides, of saccharin, of marijuana, of various promising pharmaceuticals, and, indeed, of nuclear power itself, to name just a few. Such appraisals of risk are basic to the go/no-go decisions faced by our administrative and political apparatus. They appear again and again to have been the weak side of the risk/benefit equations underlying such decisions. It is not surprising that this has been true--the investigatory task of identifying and measuring hazards can be enormously difficult.

How are we to shore up our truth-seeking capability as to the risks of the multiple use of nuclear power in space, including use by commercial enterprises? I suggest that we should do what an individual person might do when confronting a hazardous medical procedure, namely seek a second opinion. The second opinion I have in mind would not be an opinion of whether a project should be approved or not, or whether it should go or not go. There need be no such second-guessing of the approval process as performed by the current

interagency panel. To repeat, the second opinion would be confined to shoring up the truth-seeking capability in the approval process with respect to the perceived degree of risk.

Specifically, what I have in mind for this hypothetical period of greatly increased activity is the establishment of what might be called a Space Nuclear Power Systems Safety Board. In membership and operation, the board might resemble the Advisory Committee on Reactor Safeguards. Thus its members would be leaders of the scientific and technical community who have unimpeachable credentials in the disciplines primarily involved. They would serve on a part-time basis, supported by a small full-time staff. Their proceedings would be as open as national security permits, taking evidence from all interested parties, including government agencies. They would not, however, be adversary proceedings, with all their procedural complexity. The board would emphasize truth finding, not procedural legalities.

I am leaving open the question whether the findings of the board should be binding on the decision makers or merely advisory. Even in the latter case, it would be persumed that the prestige of the individual members and of the board as a whole would ensure that its findings carried important weight and that they would in virtually all cases be factored into the deliberations of the decision makers.

An important part of the activities of the board might have an international prespective. Bearing in mind that space missions launched by any country can place at risk the citizens of other countries, the board should make public all its findings, within limits of national security, and should report regularly to an appropriate United Nations body about its activities.

Establishment of such a safety board can have benefits not only on the plane of reality but also--as can sometimes be almost as important in the field of public regulation--on the plane of appearances as well. This may be particularly important regarding continued activities by the interagency panel. There is a widespread public preception that when program sponsors are evaluating risks, technological enthusiasm can overwhelm prudence. It was just such a preception that led Congress in 1974 to break up the Atomic Energy Commission (AEC) and split off its regulatory function.

I am aware that in the present interagency panel process the leading individuals in the safety review are well insulated from the mission program offices in their respective agencies. This was true in the AEC as well, however, and had little impact on public perceptions. The fact that the activities are carried on under the same rod, administratively speaking, has always been sufficient to breed the suspicion of a conflict of interest. The findings of the safety board, to the extent that they corroborate, or are not inconsistent with, those of the interagency panel, can help to dispel any such suspicions, to reinforce the decisions of the panel in individual cases, and to buttress its general standing in the community.

However, this effect on the plane of appearances will be but an
incidental benefit of the safety board, one hardly sufficient by
itself to justify its establishment. The main reason for having such
a board will continue to be found in the needed contribution it makes
to truth-seeking capability. The rationale is well expressed in the
following remarks made recently by J. G. Kemeny, president of
Dartmouth College and chairman of the President's committee on the
accident at Three Mile Island:

> We have to have a forum for effective discussion of highly
> technological issues so that there is a clear consensus
> about what science and technology say about an issue.
> Then the political process can make the value judgment.

SAFETY THROUGH TECHNICAL INTEGRITY

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I have been asked to discuss some aspects from the federal government's
point of view of what it takes to create and run a large successful
space nuclear program in which the health and safety of the public
must be a major concern.

The views I will express today are my own and do not necessarily
reflect those of any government agency. Also, I do not represent, nor
am I a spokesman for, any part of the federal government. I speak as
a private citizen who just happens to have had some nuclear experience
in the federal government. I have no experience in space technology.

The basis for my comments is 25 years of working in the field and
at the headquarters organization of the Naval Reactors Program under
the leadership of Admiral H. G. Rickover. I left that program and the
federal government in 1979, after having spent the previous 15 years
as the deputy director. That program chalked up an enviable safety
record since its beginning in 1948. Today the Navy has more than 130
nuclear-powered ships in operation that have steamed over 150 million
miles and have clocked more than 2,400 reactor years of operation
without having had any adverse effect on the health and safety of the
public or the environment. Maybe some of the lessons learned in that
program could be helpful in the program you are discussing at this
symposium.

First a word about nuclear safety as it pertains to this project
and to the public and the environment. In a project such as the one
being discussed here today, the federal government will bear the full
responsibility. The law requires that. The requirements will be more
stringent than those now imposed on other nuclear power programs,
since the consequences of a failure could be more serious and more far
reaching. If a project such as this is to come into existence, the
government will have to use methods and approaches that are departures
from the past and present but that nevertheless incorporate the
knowledge gained from those experiences, both the good and the bad.
The methods used today for evaluating nuclear safety and the means of
achieving it in the design will not suffice for this project.
Entirely new concepts bearing on nuclear safety will need to be
developed. Systems and equipment will need to work with a higher

degree of certainty. The role the federal government should play in
this endeavor is what I intend to discuss.

The success of any program of such magnitude depends upon the
totality of the program. By that I mean that you cannot take a piece
here and there, accepting the easy parts, rejecting the difficult, and
then expect it to work. There is no simple, easy way.

I am reminded of a situation that illustrates this point. A number
of years ago, Admiral Rickover and I were out in the Pacific and
talking to a senior admiral in charge of a large number of ships. He
voiced his concern that so many of his nonnuclear-powered ships were
not capable of meeting their commitments because they were always
breaking down. He had nothing but the highest praise for his
nuclear-powered ships because they invariably were fully operational.
He asked Admiral Rickover to please provide him with a copy of
Rickover's maintenance procedures so that he could distribute them to
his nonnuclear ships. He was genuinely convinced that all he needed
was the procedures so that he could distribute them to his ships and
then they would all work just fine. He did not realize that the high
reliability of the nuclear plants involved an entire spectrum of
elements, including design, manufacturing, testing, personnel,
training, inspection--just to mention a few. I will not relate the
discussion that followed, but needless to say, the Fleet Admiral did
not get the procedures. However, he did get a stern lecture. I am
afraid there are too many well-meaning people who think that there is
some cookbook approach to getting safe and reliable engineering plants.

That is why I was at first reluctant to speak on this subject here
today. It is as if I had all the answers and I am up here providing
you with a list of ten or so things to do. Then if you do these
things, everything will fall into place. Let me emphasize at the
beginning that nothing could be farther from the truth. All I can do
is to hit on a few points that, in my opinion, if they are not
covered, will lead to problems--they will not guarantee success.

First of all, a national commitment to the program is necessary.
In other words, there must be a recognized need and an established
mission for what you are attempting to produce. I am reminded of a
program that has some similarity with the one under discussion here
today--the Aircraft Nuclear Program. President Kennedy, when he ended
ANP in 1962, kept asking if we really needed it. When the answer was
no, not really, the program was dead. Make sure that there is an
indisputable need. Too often, projects get started because
contractors are looking for work. They dream up a new device or
gadget that will take years to develop and lots of money. Now they
have got to find a sponsor, and believe me, they have found some
ingenious ways of convincing the government that there is a dire need
for this thing, whatever it is. In some cases they could not care
less what it is to be used for as long as the government supports it.
If this is a description of the project being considered at this
symposium, then you are doomed to failure. You will also need support
within the administration and the Congress. It helps if the need is

so clear that its life does not depend on the results of the elections held every four years. I will not dwell on how you get this type of support, but without it, you go nowhere.

Once having decided that there is a compelling need for what you are attempting to do and you have the necessary support, you can start laying the groundwork to get it done.

Since more than one agency of the government can and will be involved, it is essential that the group running the project have the authority to act within each of the various agencies. For example, the director of the project, and for the sake of this discussion I will call him the nuclear project director, should have a position of authority within each agency. If the program is of sufficient importance, it may take an executive order to achieve this. At least, there needs to be a written memorandum of understanding between the agencies that gives him authority to act across the agencies.

Selecting a nuclear project director to head the effort then becomes the single most important issue. The person selected should be committed to remain in the job for at least 10 years—the longer the better. Of course he can be replaced if the need arises. He should have had some record of accomplishment and be an engineer—not a pure scientist or a management expert. The right person will be hard to find.

It will then be up to the nuclear project director to start assembling a technical group. This group should consist essentially of engineers. It should be a relatively small group of highly competent people who are willing to commit themselves for the long haul. In order to attract and retain the calibre of people needed, the project needs to have the same types of personnel exemptions from the requirements of the Office of Personnel Management as did the Atomic Energy Commission (AEC), the Nuclear Regulatory Commission (NRC), and the National Aeronautics and Space Agency (NASA). The group should be located in the Washington, D.C., area.

The group should be given the responsibility for all elements and phases of the program, such as research, development, testing, design, manufacture, operation, and fiscal control. However, this group should be responsible only for the reactor and its power-producing equipment, not the entire space vehicle.

A government laboratory will in all likelihood be selected to do the actual work. There are a lot of dos and don'ts involved here. The laboratory designated, or that portion of it, should be totally dedicated to this project, and to this project alone. The government and contractor managers of the laboratory should report directly and solely to the nuclear project director. The contract between the government and the laboratory contractor should be administered and controlled by the nuclear project director. Sufficient safeguards should be in the contract to prevent the contractor from using this laboratory as a place to train his people at government expense for other work. The nuclear project director should have approval authority over the management of the laboratory.

Since the product you envision is only a part of the total package, there is a need for the nuclear project director to work with those responsible for the other parts. He must be the final authority on any matter that could directly or indirectly affect reactor safety.

There will be a tendency to establish one or more so-called "oversight committees." One of them could well be a "nuclear safety oversight committee," that would supposedly resolve disputes among the various directors. Although I think that these committees are of little value if the nuclear project is handled properly, and can even be harmful, I suspect that with the current mood regarding anything nuclear, you will not be able to avoid them. They do need to be kept to an absolute minimum.

A project of this size will be expensive and will take years to complete. Because of this, there are a number of pitfalls that have to be avoided. The government must guard itself against being lulled by those who will say in the beginning that the job can be done for X millions of dollars and in Y number of years. They are expert at painting rosy but unreal pictures. They are also expert at downplaying technical problems with the old hack, "Oh! that's just a minor problem that can easily be solved." It just is not so. I am reminded of a proposal made by a large reputable nuclear design outfit not too many years ago seeking a Navy contract. One element of the design had a fluid operating at minus 300° separated by a 0.25-in. metal plate from another fluid operating at 3000°. When asked about the metallurgical and heat transfer properties of the plate, the contractor just waved it off with, "We don't think that's going to be a problem."

Big projects tend to get bigger as contractors see avenues to keep their people employed longer and to make their companies more profitable. There will be a tendency to use this project to enter into all kinds of new research and development projects. Only a strong hand by the nuclear project director can control this.

Allow me to discuss for a few moments the problems of the bureaucracy. Unfortunately, it is a fact of life you cannot avoid, but you can try to make it less stifling. Admiral Rickover stated a number of times that had he tried to get the NAUTILUS built in the 1970s, he could not have done it in less than 10-15 years and would have required at least 10 times the cost because of the increased size of the bureaucracy. A serious problem brought about by bureaucracy involves changing the mission requirements. A number of programs suffered this fate, including ANP. As the government, including the military, are continually changing people, you are subject periodically to being given new and usually more difficult mission requirements. For a program that extends over some 10 years, this can be a problem. As I understand your project, it will or could directly involve a number of government agencies: the Department of Energy (DOE), the Department of Defense (DOD), and NASA. You can also expect "help" from several others, including State, the Environmental Protection Agency (EPA), NRC, Commerce, Health and Human Services,

Interior, Labor, and Personnel Management. This is not to overlook the White House with OMB (Office of Management and Budget), and the Office of the Scientific Advisor, or Congress with its growing number of interested committees and GAO (General Accounting Office). Each of these groups feels, with some degree of justification, that it has a legal right to know what is going on, particularly if it involves its areas of interest. The task of the nuclear project director is to attempt to weave his way through this morass on a daily basis. He does this best if he has two attributes on his side. One is credibility. It is hard to come by, but it is something that can be achieved. That is why selecting the right director becomes so important. The second attribute is authority. He must have sufficient authority, within the agencies he is working with, to be able to speak and have his voice heard.

As I previously mentioned, an engineer should run the project. The vast majority of the tough problems will be engineering problems, not scientific problems. There will be a need for scientists, but they will have to be controlled; otherwise the project will be forever pursuing ideas rather than producing results. Another group that needs to be controlled, or better still, excluded, is the so-called management experts. Unfortunately, industry and the government are loaded with them, and they are all looking for a home. The success of this program will be determined by the ability to solve difficult engineering problems. It involves designing hardware that will work under all types of adverse conditions and for long periods of time.

I realize that my comments have been very critical of contractors. While this has been done on purpose, I do need to put it in better perspective. Contractors are essential. They provide the wherewithal for solving the host of technical problems that this project will face. In today's world you cannot do the job solely with government employees. What I want to stress is that from the government's point of view, the contractor's role must be controlled and channeled so that the government can be in the best position to carry out its responsibility for safety.

So far I have discussed how to set up an organization to carry out the program. Now I will discuss some basic elements that ought to be part of the design philosophy that will get you closer to achieving nuclear safety.

The design should be kept simple. Today there is a tendency to over-complicate. Safety and simplicity go hand in hand. The more complex a system gets, the more things there are that can go wrong. Furthermore, the design should be forgiving. Recognition must be given to the fact that things will go wrong. The design cannot be made so close to limits that there is no margin for error.

Allow no item or piece of equipment to be used in the final product that has not been thoroughly tested in the environment and for the length of time it will be used. The final product is no place to test any feature of the design. A prototype of the actual end product should be tested under the most severe conditions it will ever see.

Because this plant will operate without human beings directly controlling its functioning, great reliance will have to be placed on completely automated controls, and conceivably for long periods of time. Unfortunately, technology as it exists today is not capable of performing this function with the full assurance that it will work all the time, everytime. This is the area that will be the hardest technical nut to crack.

All features of the design must be as fail-safe as possible, both independently and in combination. Another laboratory or independent, technically competent group should perform a separate design review and a safety analysis of the design.

This brings into question the role NRC could or should play. While I do not advocate having such a program licensed, I do feel that having NRC in a role of reviewing the safety aspects of the design would be of immeasurable benefit. Such an arrangement would necessarily have to be carefully structured.

Nuclear power in the United States today would not win a popularity contest. Without going into the reasons why, suffice it to say that putting a nuclear power plant in space is going to be met with public resistance. The question of how the government will handle such a program with the public will have to be answered. While the technical details can be withheld for national security reasons, the very existence of the program cannot be withheld, in my opinion. You can expect to receive strong public sentiment against the program. Political or moral issues are best handled by Congress. It will be up to the federal government to handle the issue of safety.

As I started this talk, the issue was the health and safety of the public. If this device were not nuclear powered, I would not be here. All the points I have discussed so far were made with that aspect in mind.

I will end with where I started. The full responsibility of the nuclear safety aspects of this programs resides with the federal government. The current philosophy that permits this burden to be put on or shared with the contractor is specious. The only way the government can meet this requirement is by setting up a strong, technically competent group within the federal government that controls all aspects of the program.

In conclusion, I would like to quote my former boss, who made the following comment in his testimony before the Kemeny Commission after the accident at Three Mile Island:

> The problems [of reactor safety which] you face cannot be solved by specifying compliance with one or two simple procedures. Reactor safety requires adherence to a total concept wherein all elements are recognized as important and each is constantly reinforced.

This philosophy is easy to espouse but difficult to accomplish.

PROCEDURES FOR SECURING CLEARANCE
TO LAUNCH REACTORS

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ABSTRACT

The procedures for securing clearance to launch reactors into space
are the same as those for launching other radioactive materials into
space. Reactor systems must be designed to meet specified criteria.
A safety analysis report (SAR) must be prepared and submitted to the
Interagency Nuclear Safety Review Panel (INSRP). The INSRP members
and subpanel members will be available for consultation and review
throughout the program, and they will review the SARs and related data
in detail. Following review of the final SAR, the INSRP will prepare
a safety evaluation report (SER) and submit it to the heads of
agencies who, in turn, will recommend launch approval (or disapproval)
to the Office of Science and Technology Policy (OSTP). Approval can
be granted by OSTP in some instances, or the recommendation of
approval can be forwarded to the President for approval.

BACKGROUND

At the beginning of a program, such as the development of space
nuclear reactors, a few thoughts and some background information
relative to the safety requirements, analysis, review, and approval
procedures for space nuclear systems are important. It is often
considered necessary to change procedures, add more or different
groups, and "reinvent the wheel" whenever a new system is designed.
This paper provides an introduction to the procedures and requirements
that have been used successfully during the last 18 or 20 years
without "reinventing the wheel" each time.

In 1961, two SNAP-3A (Systems for Nuclear Auxiliary Power) units
were flown by the Department of Defense (DOD) and the Atomic Energy
Commission (AEC) (now Department of Energy (DOE)) on board two DOD
navigation satellites. Approval to launch these units was granted
following an informal DOD and AEC review. In 1962, no formal policies
or procedures existed to control these activities. In preparation for
the SNAP-9A launches in 1963, an expanded review group was established

and more detailed procedures were implemented. The National
Aeronautics and Space Administration (NASA) was invited to participate
in the reviews, although the launches were for DOD navigation
systems. At that time, the responsibility for these reviews was made
a part of the responsibilities of the joint AEC/NASA Space Nuclear
Power Office. At the same period, reviewers were looking at the
proposed SNAP-10A reactor system, which was launched as a test project
in 1965. It was during these early reviews and launches that
efficient and comprehensive review and approval procedures were
developed. Table 1 summarizes the space nuclear power systems
launched by the United States to date.

Since the specialists were inexperienced in working with the
space-related nuclear environments, it was critical to recognize and
allow for possible launch failures. It was also obvious that the same
procedures used for ground-based systems could not be followed,
because the systems were lightweight and could not be enclosed in big
protective containers or heavy shielding and because potential launch
failures on or near the pad and reentry following an unsuccessful
launch and short orbital lifetimes could result in the system falling
to earth in unknown and uncontrolled areas. Further, approval at the
highest level was required. It was critical for the Department of
State and the President and his staff to understand the potential
risks of these launches. The potential for political repercussions
was great in case of failure because of impact and possible fuel
release on foreign territories.

ESTABLISHMENT OF A REVIEW COMMITTEE

During this period of early reviews, representatives from AEC, DOD,
and NASA were outlining areas and procedures that could make reviews
and approvals more consistent and efficient. Concerns and
recommendations were forwarded to the heads of the agencies for
consideration. Meetings at the agency level were held, and it was
determined not to use a standing formal interagency committee, because
some details were classified, so that open, public participation was
not possible. An ad hoc group was approved. This group has since
taken on the name "Interagency Nuclear Safety Review Panel" (INSRP)
and is often referred to as the "panel." Figure 1 shows the panel as
used for the SNAP-27/Apollo review. Figure 2 lists the membership by
organization of the working groups or subpanels as appointed for the
same review of Apollo.

As part of these early reviews, an organization with experience in
and growing understanding of the many potential failure modes was
developed. Since that time, the same general procedures for
designating the review panel have been used. The user agency requests
participation of the other two, the coordinators meet and determine
support needs, and in turn, the specialists are designated by the

TABLE 1 Summary of Space Nuclear Power Systems Launched by the United States (1961-1982)

System	Spacecraft	Date	Mission	Outcome
SNAP-3A	Transit 4A	June 29, 1961	Navigational	Successfully achieved orbit.
SNAP-3A	Transit 4B	Nov. 15, 1961	Navigational	Successfully achieved orbit.
SNAP-9A	Transit-5BN-1	Sept. 28, 1963	Navigational	Successfully achieved orbit.
SNAP-9A	Transit-5BN-2	Dec. 5, 1963	Navigational	Successfully achieved orbit.
SNAP-9A	Transit-5BN-3	April 21, 1964	Navigational	Mission aborted; burned up on reentry.
SNAP-10A (Reactor)	SNAPSHOT	April 3, 1965	Experimental	Successfully achieved orbit.
SNAP-19B2	Nimbus-B-1	May 18, 1968	Meteorological	Mission aborted; heat source retrieved.
SNAP-19B3	Nimbus 111	April 14, 1969	Meteorological	Successfully achieved orbit.
SNAP-27	Apollo 12	Nov. 14, 1969	Lunar	Successfully placed on lunar surface.
SNAP-27	Apollo 13	April 11, 1970	Lunar	Mission aborted on way to moon, heat source returned to South Pacific Ocean.
SNAP-27	Apollo 14	Jan. 31, 1971	Lunar	Successfully placed on lunar surface.
SNAP-27	Apollo 15	July 26, 1971	Lunar	Successfully placed on lunar surface.
SNAP-19	Pioneer 10	March 2, 1972	Planetary	Successfully operated to Jupiter and beyond.
SNAP-27	Apollo 16	April 16, 1972	Lunar	Successfully placed on lunar surface.
Transit-RTG	Transit (TRIAD-01-1X)	Sept. 2, 1972	Navigation	Successfully achieved orbit.
SNAP-27	Apollo 17	Dec. 7, 1972	Lunar	Successfully placed on lunar surface.
SNAP-19	Pioneer 11	April 5, 1973	Planetary	Successfully operated to Jupiter and Saturn and beyond.
SNAP-19	Viking 1	Aug. 20, 1975	Mars	Successfully landed on Mars.
SNAP-19	Viking 2	Sept. 9, 1975	Mars	Successfully landed on Mars.
MHW	LES 8/9	March 14, 1976	Communications	Successfully achieved orbit.
MHW	Voyager 2	Aug. 20, 1977	Planetary	Successfully operated to Jupiter and Saturn and beyond.
MHW	Voyager	Sept. 5, 1977	Planetary	Successfully operated to Jupiter and Saturn and beyond

SOURCE: Bennett (1981).

coordinators. The coordinators also determine the requirements for subpanels to support the technical needs.

Since 1962, Mr. Thomas Kerr has been NASA's coordinator in the reviews for all space nuclear systems. Mr. George Dix was the first DOE (AEC) coordinator, from the early 1960s until he was transferred, and was succeeded by Ted Dobry and later by Marvin Norrin. DOD changes coordinators and supporting personnel on a regular basis as military transfers occur. There are individuals who have been involved for several years and there are others, of course, who have been involved in the review of one or two systems before moving on and being replaced. At the present time, many capable individuals are available and receive the necessary support from management.

The INSRP is in the early stages of reviewing NASA's Galileo system and the European Space Agency's International Solar Polar mission. Several new problem areas have to be solved for these systems. First, the systems will be carried into orbit aboard the Space Shuttle. Second, they will use the wide-body Centaur as the upper stage to take them from the shuttle to their mission environments. Third, they will have additional propulsion for trajectory shaping, guidance, and station keeping. Finally, they will use new radioisotope thermoelectric generators (RTGs). These features are not particularly unique; they are routine for space systems. Changes made in the past technologies always seem unique. In previous systems, the spacecraft and RTGs or reactor at the top of the stack were somewhat isolated by distance from the most hazardous potential environments during the launch phases. The shuttle cargo bay, however, is located in the most hazardous area possible; i.e., the Centaur is in the bay in the immediate vicinity of the RTGs. The RTGs are thermally hot and require auxiliary cooling while the shuttle doors are closed, and there are several propellant tanks in the same bay. Needless to say, we recognize the potential risks this combination provides.

What relationship does this have to reactor safety, review, and approval procedures? Reactors will not be launched in an operating mode; they will be made safe so that nothing can happen during launch short or immediate reentry; and they will be thermally cold. Further, they will be "started up" only after they have reached acceptable orbits or trajectories; thus there should be little or no concern.

However, mishaps and failures usually do occur because, for example, workers fail to follow procedures; hardware fails to function properly; people get involved; or no one thinks of some combination of events, actions, or failures that, according to "Murphy's Law," will occur. In planning applications and developing hardware it is important to involve specialists who understand the systems and at the same time can ask searching questions. These specialists should include at least one or two professional "fault finders"--individuals who are not required to respond positively to management plans or follow the party line. They must understand the systems and the potentials; they should have sufficient technical stature to be

accepted by the community; they must be supported by management to the extent that they can require changes without fear of discipline; and they must be well enough informed about the total mission that they can recognize the magnitude of potential failures. In applications such as the use of reactors or radioactive materials in space or other potentially hazardous activities of this level where failures can lead to many injuries or fatalities, a rigorous safety evaluation by the developer and user is critical.

INSRP PARTICIPATION

Sometimes users tend to assume that they can rely on the review panel to ensure safety, and although this may be true, the most probable results are added expenses and program delays. The panel participates almost from the inception of these programs. Each system is evaluated by INSRP in a three-stage review process: after submission of (1) the preliminary safety analysis report (PSAR), (2) the updated safety analysis report (USAR), and (3) final safety analysis report (FSAR). Members of the panel and subpanels are involved during critical program review meetings, in test program reviews, in working group meetings, or any time they can obtain significant information that will help them understand and evaluate the system. Figure 3 illustrates a typical development and INSRP review schedule. The INSRP involvement and review is ongoing and extends over a long period.

When INSRP personnel recognize potential difficulties, data or information deficiencies, or any other item that may lead to problems during a review, this information is given to project personnel for action. Participation during the early phases of a project does not, in any way, commit reviewing personnel to any particular position. An attempt is made to prevent inconsistencies, but if information that changes the outlook on a particular problem comes to light at a late date, the reviewers depend on the latest available data. For example, during the review of the SNAP-19/NIMBUS-B system, some of the late data indicated a possible need for a major design change in the fuel capsule, and time was short. The proposed changes could not be guaranteed until very late, just before launch, and as a result, both designs were reviewed, and two safety evaluation reports (SERs) were written using the appropriate inputs.

USEFUL STEPS BEFORE PREPARING AN SAR

So far, I have discussed the INSRP and its operations: how it functions, its history, its successes, and its independence. It was a very conscious decision to approach the procedures in this manner. There are several papers by DOE program personnel (NUS Corporation, 1974; Bennett et al., 1980a,b; Bennett, 1981; Bennett et al., 1981) that cover various aspects of the space nuclear systems and their

review. Note in particular Bennett's (1981) article, "Overview of the U.S. Flight Safety Process for Space Nuclear Power."

The first and perhaps most critical requirement for users is to communicate with the appropriate INSRP coordinator. As a result of this contact, the experience and information collected by the INSRP is made available. The data requirements can be discussed, guidance concerning SAR requirements is given, and INSRP personnel provide ongoing review of the activities.

Second, the Overall Safety Manual (OSM) (NUS Corporation, 1974) should be obtained, reviewed, and understood. This suggests study of the basic documentation requirements to obtain an understanding of the types of data that are in the OSM by surveying the table of contents, the references, the technical models, and the basic data that are included. It is important to recognize the method used in putting the information together to study similar systems' SARs so that there will be no surprises, and to ask questions. The OSM volumes provide background information concerning demography, oceanography, and meteorology, and much source material common to all systems. In addition, they provide an outline and guidance for preparing SARs. The contents of SARs include the information listed in Table 2.

It is certainly not recommended that one should obtain an OSM and begin writing a SAR in a "vacuum." There are many aspects of a launch and launch-related information that could be extremely expensive to duplicate and that have already been documented. There is extensive experience in areas such as calculated and historical values of failure rates. Each launch vehicle is evaluated the first time it is used for one of these systems, and in follow-on uses the differences are documented. Launch site characteristics are considered consistent for the same area from launch to launch except for seasonal variations or specific modifications. These consistent characteristics are included in the OSM reference data and other source documents that are used with the OSM.

Third, at an early date, it is important to invite the INSRP coordinators and members, as determined necessary by the coordinators, to meet with project personnel to exchange information and discuss operational and review philosophy. This allows individuals to meet the principal contacts they will be working with during the next several years. Many of the unknowns can be discussed and the tendency to consider each other the opposition can, to a large extent, be eliminated. The INSRP is not interested in preventing a launch; it is interested in evaluating the potential for mishaps and avoiding or minimizing injury or death of people.

CONTENTS OF SAFETY ANALYSIS REPORTS

According to Bennett's (1981) article, in general, the Safety Analysis Reports consider the following types of accident environments (categorized by mission phase):

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TABLE 2   Contents of Safety Analysis Reports

Reference Design Document
Description of:
Mission and flight system summary
Nuclear power source (including type of fuel, design requirements, materials and their
  properties, radiation properties, power conversion subsystem, and ground support
  equipment)
Spacecraft (including location and attachment of the nuclear power source)
Mission profile
Launch vehicle (including flight safety and tracking)
Reference trajectory and flight characteristics (including launch conditions)
Launch site (including demographic, topographic, and meteorological characteristics)
Range and radiological safety

Accident Model Document
Description of:
Summary of mission and flight system
Accident and radiological models and data (including test data that support the
  analysis)
Vehicle and nuclear power source failure mode analysis (including prelaunch, launch,
  ascent, and space operation, with a description of the potential accident environment
  and flight contingency options)
Nuclear power source response to accident environments (including prelaunch, launch,
  ascent, reentry, breakup, impact, and postimpact--both land and water)
Mission failure evaluation (includes accident probabilities and quantity of radioactive
  material potentially released)

Nuclear Risk Analysis Document
A probabilistic description of the potential radiological risk to the world's population
resulting from potential accidents which could involve the nuclear power source of a
spacecraft.

SOURCE: Bennett (1981, p. 431).

Prelaunch, Launch, and Ascent Phases:
  --Explosion overpressure.
  --Projectile impact.
  --Land or water impact.
  --Liquid propellant fire.
  --Solid propellant fire.
  --Sequential combination of the above.
Orbit and/or Flight Trajectory Phases:
  --Reentry.
  --Land or water impact.
  --Postimpact environment (land or water).

On-orbit contingency options (including retrieval) are considered as
appropriate.

The safety and safeguard requirements during development,
fabrication, testing, and transportation of the nuclear power system,
prior to the arrival of the hardware at the launch site, are provided
and enforced by the usual DOE operational requirements and orders and
include specific guides at all levels, such as surveys, and reviews
and approvals of facilities, procedures, personnel, equipment, and
emergency plans. Transportation and storage of special nuclear
material are always taken seriously. SARs consider the storage and
handling prior to and during installation of the nuclear power supply
on the spacecraft and all phases following installation. Detailed
safety tests are required, and they are evaluated as part of the
analyses associated with the SAR and review.

INTERAGENCY NUCLEAR SAFETY REVIEW PANEL

An outgrowth of the early decisions to review space nuclear power
systems by a special interagency committee, this procedure has now
become policy. The exact makeup and procedures have not been
formalized to the point that it is a "one, two, three or follow a
specific guideline and everything will be fine" activity. The
President has directed that these systems be reviewed using the ". . .
Interagency Nuclear Safety Review Panel consisting of members from the
Department of Defense, Department of Energy, and National Aeronautics
and Space Administration . . .." The Nuclear Regulatory Commission is
also to be requested to participate as an observer when appropriate.
The procedures, personnel, organization, and other details are left to
the discretion of the agencies and the INSRP, as appointed.

Figure 4 illustrates the involvement and path for obtaining
approval to launch these systems. There are many organizations,
meetings, studies, tests, and development activities and phases that
feed into the four blocks on the left. Also, many technical and
"political" questions have to be analyzed and answered. Data flow

from many related organizations. Vehicle and launch environment parameters are examined from a different point of view than is normally necessary for operations. Environmental factors may be critical in the event of an abort, and special studies are conducted. Subpanels of the INSRP, as indicated in Figures 1 and 2, are charged with the responsibility of evaluating the many likely or reasonably probable events that could occur during or following an abort and that could possibly release radioactive material. This will include the immediate as well as long-term effects. The input for these analyses often requires extensive and expensive test programs. Some of these tests are funded by a particular spacecraft office or nuclear power program office if the effect is unique. In other instances, there are ongoing tests to accumulate information over periods of many years. These are funded separately as part of a long-term safety program. The box showing the INSRP is deceptive from a time requirement point of view. The INSRP becomes involved during the early phases, almost during the concept period. The INSRP certainly influences the documentation requirements and the schedule for these documents. Normally there are at least three formal INSRP review meetings: one for the PSAR, one for the USAR, and one for the FSAR. Other meetings are held as required. The INSRP subpanels meet as often as necessary to guide program personnel in the development of needed information. Although the subpanels do not develop the data, they analyze them, perform calculations and tests as needed, recommend areas for further analyses of tests, and provide experience and assistance to help program personnel understand the safety needs and avoid unnecessary costs whenever possible.

The INSRP and its subpanels are not made up of three duplicate specialty groups. Experts are appointed as needed and are available from whenever they work. In some instances each agency has the necessary capability, while in others one agency may have the greater capability in house or by contract. The INSRP members are government employees. The subpanels may have government or contractor members. In addition, each panel and subpanel member can have as many technical advisors from his or her own agency, contractors, or other agencies as considered necessary.

SAFETY EVALUATION REPORT

Once the FSAR is reviewed and the INSRP members are satisfied, the panel prepare a safety evaluation report (SER) and submits it to the agency heads for review and approval. This is the document that provides necessary summary information to the offices indicated on the right-hand side of Figure 4. Although the INSRP does not make a recommendation of launch approval or disapproval, the results of the INSRP review, as documented in the SER, certainly lead managers to a position where they can make the decision. A further reason to expect most SERs to lead toward launch approval is the recognition by program

personnel, throughout the development and fabrication phases, that if
the INSRP or its subpanels find potentially serious design problems
and point them out, it is prudent to listen and make the necessary
changes before getting to the approval stage. This has been done
several times in the past.

REQUEST FOR LAUNCH APPROVAL

After the SER has gone to the agency heads and they are satisfied with
the findings, the two supporting agency heads submit letters of
concurrence to the user agency, and in turn, the user submits a letter
to the Office of Science and Technology Policy requesting launch
approval. This request may be acted upon at that level in some
instances, and in others, the approval is granted through the Office
of the President.

When more than one system will be launched over a reasonable period
of time using the same facilities and launching vehicle types, it is
not unusual to have the complete series approved at one time by the
President. Notification prior to each launch is required so that
proper contingency preparation can be in place during the launch and
critical periods.

CONCLUSIONS

The review procedures as developed and implemented by the INSRP have
been successful and well accepted by all levels of government. These
procedures have been presented to and accepted by the Scientific and
Technical Subcommittee of the United Nations Committee on the Peaceful
Uses of Outer Space. The United States supports a rigorous space
nuclear safety program that provides for testing and analyses of power
sources intended for space applications. The SER includes a
probabilistic risk analysis technique to assist decision makers in
assessing the risk versus benefit from the use of the nuclear power
system. The INSRP does not have any line authority or directive power
over programs using nuclear power systems; however, its reviews
presented as negative findings certainly would influence management.
As a result, from a practical point of view, program management
listens to the INSRP and considers all items that are marginal or
questionable during all phases of a program.

To my knowledge, there are no indications that a new procedure or
policy will be implemented in the immediate future. The INSRP has
participated at a minimal level, but has nevertheless been involved,
in the planning and discussions concerning future reactor programs and
possible needs for reactors in space. The procedures for these
activities are the same as the ones that have been used for several
years and that are being used for the Galileo and International Solar
Polar programs.

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FIGURE 1   Interagency Nuclear Safety Review Panel (Apollo missions).

RANGE SAFETY

(PRELAUNCH, LAUNCH, EARLY ASCENT) DEFINE ABORT ENVIRONMENT AND FUEL RESPONSE SOURCE TERM.

CHAIRMAN - DoD - USAF/IG

NASA/MSFC	USPHS (EPA)
NASA/KSC	SANDIA CORP
NASA/LeRC	AEC (DOE)/ALO
DoD/ETR	AEC/REG (NRC)
DoD/AFSC	AEC (DOE)/SNS

REENTRY

(SUBORBITAL, ORBITAL, SUPERORBITAL EXPLOSION) DEFINE REENTRY ENVIRONMENTS, FUEL RESPONSES, SOURCE TERM.

CHAIRMAN - NASA/ARC

NASA/LeRC	AEROSPACE CORP
NASA/MSC (JSC)	DoD/USAF IG
NASA/MSFC	AEC (DOE)/SNS
SANDIA CORP	AEC/REG (NRC)

METEOROLOGY

EVALUATE ATMOSPHERIC MODELS, LOCAL AND WORLDWIDE WEATHER, DETERMINE SOURCE DIFFUSION, AIR AND GROUND CONCENTRATIONS DOWNWIND.

CHAIRMAN - AEC (DOE)/DBM

ESSA (NOAA)	AEC (DOE)/SNS
SANDIA CORP	USPHS (EPA)
BATTELLE	AEC/REG (NRC)
NASA/MSFC	AEC (DOE)/DBM
DoD/AFWL	AEC (DOE)/HASL

OCEANOGRAPHY

EVALUATE SHORT/LONG TERM EFFECTS OF MARINE RELEASES ON BIOTA, UPTAKE, FOOD, FRESH WATER.

CHAIRMAN - AEC (DOE)/DBM

U. OF WASH	JOHNS HOPKINS
USNRDL	DoD/AFTG
AEC (DOE)/HASL	AEC/REG (NRC)
AEC (DOE)/SNS	BATTELLE
MARINE RESOURCES COUNCIL	

BIOMEDICAL

EVALUATE BIOMEDICAL MODELS OF RESPONSES OF INDIVIDUALS AND ANIMALS TO INGESTION OF PLUTONIUM AS THE RESULT OF POSSIBLE EXPOSURES DURING ABORT CONDITIONS.

CHAIRMAN - AEC (DOE)/DBM

NYU	BNWL
ORNL	LOVELACE
U. OF ROCH.	LRL
LASL (LANL)	AEC (DOE)/DBM

FIGURE 2 Working groups (Apollo missions).

FIGURE 3 Typical flow of space power systems development and review activities.

FIGURE 4 Safety review and launch approval process.

OUTSTANDING RESEARCH AND DEVELOPMENT ISSUES

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STATUS OF HIGH-TEMPERATURE HEAT PIPE TECHNOLOGY

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ABSTRACT

This paper discusses the application of heat pipes to nuclear reactor
space power systems. Characteristics of the device that favor such an
application are described, and recent results of current technology
development programs are presented. Research areas that will need to
be addressed in demonstrating that adequate lifetimes can be achieved
with evaporation/condensation cycles operating at high temperatures in
a reactor environment are also discussed.

INTRODUCTION

Since it first emerged nearly two decades ago as a product of the
national space power program (Grover et al., 1964), the heat pipe has
been undergoing a gradual transition from a potentially useful
laboratory curiosity to a commercial unit used in a considerable
variety of applications. The pace of development and application has
at times been quite modest, but now the device has been produced in
quantities measured in millions, and prospects for further expansion
of its areas of utilization are excellent.

Among the more important applications have been its use to
stabilize the permafrost around the stanchions supporting the
above-ground section of the Alaska pipeline (Waters et al., 1975), its
use for heat recovery in heating and air conditioning systems as well
as for heat recuperation in industrial furnaces and ovens (Ruch,
1975), and its adaptation to waste heat boiler systems (Littwin and
McCurley, 1981). Recently, over one million units have been built for
just one application--the cooling of an audio amplifier, which is
produced abroad in large quantities (Osakabe et al., 1981).
Increasing use for

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heat radiation and temperature control systems in satellites has occurred in both the United States and other countries (Arakelov et al., 1976; Harwell and McIntosh, 1981; Kelly and Reisenweber, 1981; Kimura et al., 1981; Ollendorf et al., 1981).

Future applications now in an active development stage include underground power cable cooling (Iwata et al., 1981), temperature smoothing in processing furnaces (Kamotani, 1981), tapping of geothermal reservoirs (Cannaviello et al., 1981), and transfer of solar heat from rooftop collectors to below-ground storage (Nasonov and Bondarenko, 1980).

Most of these applications are for heat pipes operating at low to moderate temperatures. Higher temperature commercial applications have been slow to appear, primarily because of the fact that liquid metals become the working fluid of choice above about 750 K. This leads to a perceived increase in hazard and a real increase in fabrication cost. As a result, the main driver for research on high-temperature heat pipes remains now, as it was at the beginning, the potential application to nuclear reactor power systems for space. The high-temperature heat pipe was invented to solve a heat transfer problem in such a system and is now a key component in the baseline design of the SP-100 space reactor system.

HEAT PIPES AND SPACE REACTOR DESIGN CONSIDERATIONS

The heat pipe has a number of attributes that make it attractive as a space reactor component. Key among these are its very high heat transport capability in a relatively simple, self-operating system and its capability of very high temperature operation. The first of these permits the design of reactor cores with many independent coolant "loops," not all of which need be operable. Thus single-point failure mechanisms can be avoided in the primary reactor coolant system, and this in turn makes possible the design of a complete reactor power system devoid of such failure mechanisms. This characteristic is important in promoting the high reliability that will be necessary for systems that must run unattended for lifetimes of 5-10 years.

The multiplicity of independent, self-contained coolant loops afforded by heat pipes has other implications for reactor design. For instance, it obviates the need for a reactor pressure vessel, and this not only saves some weight but also permits the design of a system that can break up easily during reentry from earth orbit. The self-operating nature of the heat pipe not only eliminates the need for valves and pumps or compressors in the primary cooling loop but also makes it possible to have automatic removal of reactor core afterheat if the reactor is shut down and all power is lost.

High-temperature operation is another important attribute of the heat pipe because the design of compact, high-specific-power reactor systems requires it. This is a direct consequence of the need to dump large quantities of waste heat by radiation. The heat pipe's ability

to start up directly from a frozen state precludes any necessity for prewarming the coolant such as is needed for liquid-metal-cooled systems.

There is one other attribute of the heat pipe that becomes particularly important for direct conversion methods of transforming reactor heat to electricity. This is the very low temperature drop that exists even at high power levels. Thus all the heat is delivered to the conversion devices at what is essentially the reactor exit temperature, permitting their operation at maximum design efficiency. Similarly, in systems where reactor heat is transferred to the conversion system by radiation, it is important that the radiating surface be essentially isothermal (at the reactor exit temperature), a state not achievable with gas or liquid-metal cooling because of the inevitable temperature drop that occurs in circulating single-phase fluid loops performing the task.

SPECIFIC HEAT PIPE REACTOR SYSTEM CONCEPTS

Some specific ways in which the heat pipe influences space reactor design and the performance demands that such reactors place on the neat pipe are shown in Figures 1-4 and Table 1. Figure 1 shows the baseline reactor design for the 100-kW(e) SP-100 space power system. Table 1 lists the performance requirements this design places upon the core heat pipes. Figure 2 shows the current SP-100 system, and Figures 3 and 4 show early configurations of 100-kW(e) power systems.

The reactor in Figure 1 is a beryllium-reflected array of 120 fuel modules that can produce a total power of 1,600 kW(t). Each fuel module consists of a stack of Mo-13Re fins and UO_2 fuel wafers, built around Mo-13Re heat pipe with lithium working fluid. The heat pipes are of arterial design, and, with an external diameter of 17.5 mm, each has the capability of carrying 17.4 kW(t) for long-term operation. A design margin of 25 percent gives the limiting value of the performance of these pipes of 21.9 kW(t). The performance requirement on each heat pipe is kept constant, despite the usual radial drop in fuel power density, by increasing the amount of fuel surrounding the heat pipes for each successive circular row of fuel modules. Table 1 shows the axial and radial power density requirements on the core heat pipes for both normal operating conditions (13.3 kW(t) per heat pipe) and the contingency case in which a heat pipe has failed and the adjacent heat pipes must take away up to 25 percent of the heat the failed heat pipe had been removing.

The manner in which the SP-100 system heat pipes deliver reactor heat to the thermoelectric conversion system is shown in Figure 2. The heat pipes emerge from the core and traverse the reactor shield (making two near-90° bends in the process to prevent neutron streaming) and then extend another 6.7 m to form a cylindrical array that radiates at 1495 K to thermoelectric panels that surround the

TABLE 1 SP-100 Heat Pipe Operational Requirements

Temperature (K)	1500
Axial power density (kW/cm^2)	
Normal operation	8.0
Contingency operation[a]	10.5
Maximum radial power density (W/cm)	
Evaporator	105
Condenser	6.5
Lifetime	7 yr in normal mode, 0.98 reliability
	7 yr in contingency mode, 0.96 reliability

[a]17.4 kW per heat pipe for a 120-heat-pipe, 1,600-kW(t) reactor.

heat pipe bank. Waste heat in this configuration is radiated directly from the outer surfaces of the thermoelectric panels.

Figure 3 shows a different type of heat pipe reactor power system, with Brayton-cycle conversion. One of the distinguishing features of the design is that the heat pipe array supplies heat to two independent Brayton loops, thus achieving a twofold redundancy. For this design, the factor limiting the amount of redundancy was the increase in total mass of the conversion units when many small-capacity units are used in place of higher-capacity ones.

The use of heat pipes to achieve redundancy in a large, lightweight, high-temperature radiator for disposing of waste heat from the power conversion units is shown in Figure 4. This system was a forerunner of the SP-100 and used potassium/titanium heat pipes to remove the heat from compact thermoelectric conversion modules. An expanded view of a radiator panel is shown in Figure 5.

HIGH-TEMPERATURE HEAT PIPE PERFORMANCE

The performance requirements for the SP-100 reactor core heat pipes given in Table 1 are demanding. That they are nevertheless realistic has been demonstrated by analyses that have been performed with the computer code HTPIPE and substantiated by experiment.

HTPIPE is an analytical model of heat pipe operation based on Cotter's (1965) pioneering work on the engineering theory of the heat pipe. The flow regimes treated by this model are illustrated in Figure 6. The code integrates the flow with mass addition regime of the evaporator with the transition vapor flow behavior of the adiabatic regime and mass depletion flow of the condenser. Inertial

recovery effects caused by deceleration of vapor in the condenser are included. The modeling of the return liquid flow includes viscous losses in the arterial passages and also any losses that may occur in distribution wicks. The model predicts performance curves of the general shape shown in Figure 7. The sonic limit occurs when the vapor velocity at the entrance to the condenser reaches sonic velocity. Wicking limits occur when the forces required to drive the fluid down the pipe as vapor and back through the wick as fluid exceed the available capillary force in the evaporator. For liquid-metal heat pipes with compound wicks, the liquid pressure drop limit is rarely a factor. The boiling limit has been observed in high-temperature sodium heat pipes, but is much less important for lithium heat pipes operating in the temperature range of the SP-100 reactor design.

The general capabilities of HTPIPE are summarized in Table 2. Its predictive capability has been benchmarked against many experimental observations. An example is shown in Figure 8 for a 2-m-long sodium heat pipe. In general, it is wall-friction factors that have been refined in the benchmarking process.

The HTPIPE code has been very useful in making design decisions for the SP-100 reactor. An example is the comparative merits of sodium and lithium working fluids as a function of temperature. Such a comparison is shown in Figure 9 for a 2-m-long, annular artery heat pipe having a 14.1-mm inside diameter. The annular artery configuration is illustrated in Figure 10, along with the butterfly artery and segmented annulus wick configurations. It can be seen in Figure 9 that for temperatures below 1500 K the low vapor pressure of lithium causes its performance capability to be less than that of sodium. Increasing the temperature above this value shows that very impressive amounts of heat can be made to flow down a channel that is only 1.6 cm^2 in area.

A program of heat pipe fabrication development and performance testing is being carried out at the Los Alamos National Laboratory to verify that heat pipes with the high-performance capability predicted by HTPIPE can be built. A recent test of a 2-m-long lithium heat pipe with a butterfly artery wick configuration (shown in Figure 10) was conducted at a heat transport rate of 15 kW at a temperature of 1500 K for 100 hours. The test point is shown in relation to the predicted performance curve in Figure 11. This performance value is displayed with those of earlier 2-m-long lithium heat pipes in Figure 12 in comparison with axial heat flux performance values required for the SP-100 reactor core heat pipes. The latter are, however, 8 m long.

Limitations of the radial heat flux that can be brought in through the heat pipe wall result primarily from the boiling of working fluid in the wick. This limitation, though important for sodium, is not expected for lithium. This expectation is confirmed in Figure 13, which shows that several lithium heat pipes have comfortably exceeded the radial flux values required in the SP-100 reactor core. A heat pipe constructed for regulating the temperature of a fuel irradiation

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TABLE 2   HTPIPE Program Analysis Capabilities

Prediction of heat pipe performance limits
  Sonic limit
  Wicking limit
  Entrainment limit

Model features
  Two-dimensional laminar flow in heat pipe evaporator
  One-dimensional compressible flow witn friction in adiabatic section
  Analysis of various artery geometries
    Wickless gravity assist
    Circular artery
    Grooved wall artery
    Helical gutter artery
    Homogeneous wick
    Screened tube
  Effect of bends on wall-friction factor considered

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capsule (labeled "EBR-II Test Heat Pipe" in Figure 13) has exceeded
tne contingency operation value for radial heat flux by a factor of 3.

The SP-100 design calls for lithium heat pipes that are 8 m in
length. A program to fabricate prototype units of these heat pipes is
currently under way. In one case, the capillary pore diameter of the
pumping wick is 30 m and the heat pipe diameter is 15.9 mm. In the
other case, the diameter is relaxed to 17.5 mm--at the cost of some
increase in the SP-100 system mass--and the capillary pore diameter is
45 m, easily attainable with a 400 mesh molybdenum-rhenium screen
that has recently become available. Both can meet the long-term
contingency operating capability with a 1.20 design margin. However,
the larger diameter is favored because its required operating regime
is farther from tne sonic limit curve, and hence the temperature drop
down the length of the pipe for its design operating condition is
less. This effect is shown in Figure 15, where temperature drop for
an 8-m-long heat pipe carrying 15 kW at an evaporator exit temperature
of 1500 K is plotted as a function of outside heat pipe diameter. A
constant effective heat pipe wall and wick thickness of 1.15 mm is
assumed in this calculation.

Although the SP-100 design requires heat pipes only for primary
transport of reactor neat, earlier reactor system designs required
lower temperature heat pipes to form the waste heat radiator surface,
as shown in Figures 4 and 5. Experimental investigations of
titanium/potassium heat pipes for this application include testing of
a 5.5-m-long dual-artery unit at the projected operating temperature
of 770 K and also under conditions of fractional capacity operation.

Figure 16 shows the temperature profiles obtained. Of particular interest are tne observations that start-up of such a long heat pipe with solid potassium in the wick presented no difficulties and operation at fractional power with as little as half the heat pipe operating--and the otner half with frozen working fluid--was stable.

HEAT PIPE LIFETIME

The heat pipe analytical and experimental work done to date nas established a very nign degree of confidence in meeting tne SP-100 core heat pipe performance requirements. Demonstrating this for an 8-m-long neat pipe is a near-term goal of the SP-100 program. The other main question about the core heat pipes concerns their benavior over multiyear operating times. Program emphasis is now shifting to demonstrating sucn lifetime capability at full operating power.

Earlier testing by a number of research groups has shown that very long operating lifetimes are attainable. It has also revealed design weaknesses that must be avoided. Table 3 lists long-term test results for litnium heat pipes fabricated from a variety of materials. Lifetimes of 10,000 hours were obtained for TZM heat pipes, but in each of three cases, failure occurred as a result of the cracking of end cap welds. Thus weld design and weld quality are of great importance, particularly for materials like TZM and high-purity molybdenum, for whicn even electron-beam welds are brittle. The W-26Re test ran without incident for 10,000 hours.

The longest litnium heat pipe lifetime demonstrated to date was achieved with a short unit fabricated from low-carbon, arc-case (LCAC) molybdenum that operated for 25,400 hours at 1700 K with a heat transport rate of approximately 1 kW. The wall of this heat pipe experienced extensive grain growth, to the point that the grain diameter became several times the 1-mm wall thickness. The unit failed when lithium leaked from several holes in the evaporator, not long after tne unit had been placed in a lathe for remachining of a support stem.

The tests listed in Table 3 all operated at 200-375 K nigher than the normal operating temperature of the SP-100 reactor core heat pipes. Thus tne test results are felt to be very encouraging in terms of projecting longevity for operation at 1500 K, particularly if weld emorittlement can be avoided and wall material grain growth kept within bounds.

MATERIALS AND ENVIRONMENT CONSIDERATIONS

The selection of materials from wnicn to fabricate the SP-100 reactor core heat pipes is severely limited by three requirements. Tne first is chemical compatibility witn the working fluid and reactor fuel. Another is creep strength at 1500 K, and tne third is ductility at

TABLE 3   Lithium Heat Pipe Life Test Data

Material	Temperature (K)	Operation Time (n)	Reference	Remarks
W-26Re	1875	10,000	Busse (1971)	
TZM	1775	10,526	Rouklove (1868)	Weld cracking
TZM	1775	10,400	Eastman (1969)	Weld cracking
TZM	1775	9,800	Eastman (1969)	Weld cracking
Nb-1Zr	1775	9,000	Busse (1966)	
Mo(LCAC)	1700	25,400		Large-grain formation

launch temperatures, which could conceivably be as low as 200 K for boost of a space power supply from low-level to high-level earth orbit. The strength requirement effectively excludes all but refractory metals from consideration. It also excludes niobium from the group of potential refractory metals. Tantalum, and niobium as well, tend to take oxygen from UO_2 fuel, potentially resulting in free uranium formation. The selection, then, is limited to tungsten, rhenium, and molybdenum or alloys thereof. Because tungsten and rhenium have high densities and are difficult to work (rhenium because it work-hardens so readily), and because tungsten has a high ductile-brittle transition temperature and rhenium is expensive, neither material by itself appears suitable. This leaves molybdenum or molybdenum-base alloys as the best materials for core heat pipe fabrication. The material of choice is an alloy of molybdenum and rhenium containing 13 percent by weight of the latter. This alloy is ductile at temperatures well below 200 K and at this temperature shows e   llent weld ductility.

Because the evaporators of the SP-100 heat pipes form part of the reactor core, the materials questions become considerably intensified owing to the effect of the reactor environment. The proximity of UO_2 fuel and lithium working fluid is illustrated by the fuel module shown in Figure 17. Part of the UO_2 abuts the heat pipe wall, which is 0.75 mm thick. The inner wall of the heat pipe is, of course, saturated with lithium. Direct contact will also occur between the UO_2 and the Mo-13Re fins. During normal operation, the fin temperature will be 1510 K adjacent to the heat pipe and as high as 1730 K in the module corners. These values, as well as other fuel conditions, are listed in Table 4.

An example of the effect of reactor environment is the diffusion of oxygen through the heat pipe wall. This is an important consideration because lithium metal will take oxygen away from even stoichiometric UO_2. During the SP-100 reactor lifetime of 7 years, about 3.5

TABLE 4 Fuel Conditions for 1,600-kW SP-100 Reactor

Fin temperature range
Normal operation	1510-1730 K
Failed interior heat pipe	1530-2150 K
Transient peak	2330 K (24 h)

UO_2 tile midplane temperature   Add 30 K to fin temperature

Total burnup (7 yr)
Average	8×10^{20} fis/cm^3, 3.5% U atoms
Maximum	1×10^{21} fis/cm^3, 4.0% U atoms

Fission rate
Average	3.6×10^{12} fis/cm^3 (115 W/cm^3)

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percent of the uranium atoms are fissioned, and because the fission products are less oxidizable than uranium, the UO_2 becomes somewhat hyperstoichiometric. This results in a freer supply of oxygen to diffuse to the lithium so that if the O_2 diffusion rate in Mo-13Re were too high, sufficient LiO_2 could form in the heat pipe to hinder its operation by blocking the flow passages in the wick with a nonvolatile liquid, decreasing the wettability of the wick structure, or decreasing the surface tension of the lithium. (A preliminary experiment to estimate the magnitude of the UO_2 diffusion coefficient in Mo-4Re and molybdenum has indicated that it is sufficiently low to prevent serious lifetime-limiting effects, but must be confirmed by more rigorous experimentation and testing.)

Other effects of the reactor environment include stresses caused by fuel swelling, the effect of such stresses on grain growth, and the effect of grain growth on heat pipe wall integrity. Fast-neutron-flux effects on grain growth and creep strength may be important. A small amount of hydrogen and helium generation, primarily from neutron capture by the ^7Li working fluid, will occur in the heat pipes, potentially modifying their behavior, and fission products generated in the UO_2 may interact with the Mo-13Re fin and heat pipe wall material.

RESEARCH REQUIREMENTS

At the present stage of high-temperature heat pipe development, it is safe to assume that the probability is very good that lithium heat pipes meeting the requirements of the SP-100 reactor design, including the lifetime requirement, can be built. The goal of the core heat

pipe development program over the next few years is to prove that this
is the case through an extensive fabrication and testing effort
closely supported by corollary experimentation and analysis. Specific
areas where research is needed are analytical modeling of the heat
pipe, characterization of heat pipe envelope materials, interactions
of oxygen and fission products with heat pipe materials, and
definition of impurity effects on heat pipe operation.

Analytical Modeling

The heat pipe code HTPIPE is able to predict steady state heat pipe
performance for a three-region heat pipe (evaporator, adiabatic
section, and condenser) where the axial dependence of the evaporator
heat input and condenser heat outflow is constant. The code does not
predict kinetic effects such as start-up behavior or the effects of
power changes. It also is not able to predict the curves shown in
Figure 16 where the heat pipe is operating over only a fraction of its
length. These are two areas where the analytical code needs further
development.

More basic work that is needed involves two-dimensional modeling of
the vapor dynamics in all three regions of the heat pipe, including
radial transfer of energy, flow profiles, and effects leading to
transition from laminar to turbulent flow conditions, particularly in
the adiabatic region. A better understanding of the energy exchange
at vapor-liquid interfaces would also be valuable.

Thermochemical modeling has been initiated in the heat pipe
development activities at Los Alamos, but this work needs to be
expanded and coupled to kinetic theory to study solubility kinetics
and material transport mechanisms to determine how these mechanisms
are influenced by the presence of impurities, and also to investigate
the effect of impurities on heat pipe performance. Performance
effects can occur because of changes in the interfacial wetting forces
caused by oxide layer formation and changes in the surface tension of
the working fluid because of dissolved impurities. Other potential
effects on performance include plugging of the wick flow channels,
increase of capillary pumping pore size by erosion/corrosion effects,
noncondensible gas generation, and the formation and accumulation of
low-volatility fluid (such as lithium suboxides). Also very important
in the analysis of impurity effects is the understanding of the
gettering behavior of materials put in the heat pipe to immobilize
impurities.

Heat Pipe Performance Experiments

Experimental investigation of high-temperature heat pipes has been
hampered by the difficulty of making measurements other than heat
transport and temperature profiles. These suffice for many purposes
but are not sufficient for investigating impurity effects, start-up

effects, material migration, and such straightforward things as the
location of excess working fluid for various operating conditions.
Radioactive tracers combined with gamma scanning techniques have been
proposed for measuring material migration rates, and currently neutron
sensing of ^{6}Li is being studied for measuring working fluid
distribution.

Observations of impurity movement and determination of the specific
mechanisms by which these materials limit heat pipe performance are
very lightly studied areas. When high-temperature heat pipes are
taken to a limit, a rather wide range of behavior is observed. The
boiling limit effect is dramatic and easily identified because the
entire evaporator begins to heat up rapidly when this limit is
reached. But for other "failures," the observed effect is often no
more than the appearance of a hot spot in the evaporator wall. A
strict mechanistic interpretation of the nature of these "hot spots,"
how they are formed, why they form in a variety of locations, and how
they grow with increasing heat input or with time at constant heat
input, does not exist. A program of research to understand their
causes and behavior could provide a valuable contribution to the
understanding of the more subtle aspects of heat pipe behavior.

Materials Investigation

Because the physical and mechanical properties data base of the
material selected for the heat pipe wall and wick of the SP-100
reactor core heat pipes, Mo-13Re, is small, a considerable amount of
work must be done to characterize this material more fully. The
needed work includes measurements of high-temperature strength,
including long-term creep, as well as measurements of low-temperature
ductility. The latter needs to be done for wrought material and
recrystallized material, with particular attention to weld ductility
at low temperature. The effect of impurities on the properties of the
material also needs to be studied in order that realistic purity
specifications can be established for obtaining the material in large
quantities. Measurements even of ordinary properties like thermal
expansion coefficients, thermal conductivity, and specific heat
measurements must be extended over the entire range of projected
operation of the material. Grain growth as a function of temperature
and stress is another important phenomenon requiring study.

As indicated earlier, the understanding of oxygen diffusion in the
heat pipe wall is very important. Hence careful measurements of
oxygen diffusion coefficients and solubility in Mo-13Re must be made
and the effects of impurities on these quantities determined.
Diffusion of specific fission product species such as iodine should
also be studied to see if any problems might arise from this source.

Studies of the effects of neutron irradiation and fission fragment
implantation on the properties of the heat pipe wall and fin module
will require irradiation testing. A program of such testing has been

initiated to investigate nuclear fuel behavior in the SP-100 fuel module geometry. These tests will use the same materials selected for the SP-100 reactor and will actually feature temperature control by means of a gas-filled lithium heat pipe. Hence they will yield important information on the effects of the reactor environment on heat pipe operation, including fission product and oxygen diffusion, neutron damage effects, and grain growth under irradiation and stress conditions.

The materials research activities outlined above have been described in terms of one alloy, Mo-13Re, because sufficient data are available on this material to justify it as the prime choice for the SP-100 reactor fuel modules. However, it is important that alternative materials be available, and hence research of similar nature should proceed on tungsten, molybdenum, and rhenium ternary alloys as well as tungsten-rhenium binary alloys and possibly others.

SUMMARY

The intent of this paper has been to show first of all that some very impressive performance and long lifetimes have been obtained in developing high-temperature heat pipes for application to space nuclear reactor power systems in general and specifically to the SP-100 nuclear reactor power system. The high-temperature heat pipes constitute extremely promising technology for the building of new space reactor systems. At the same time, it must be recognized that extensive testing and supporting research need to be done to prove that high-temperature heat pipes can operate at high power, with high reliability, for lifetimes exceeding 7 years. Such a program is now getting under way with a goal of developing prototypical units with demonstrated long-lifetime capability within 4 years and proving that the required reliability exists within another 2. The latter program will use Bayesian testing methods to establish reliability rather than relying on purely statistical test methods.

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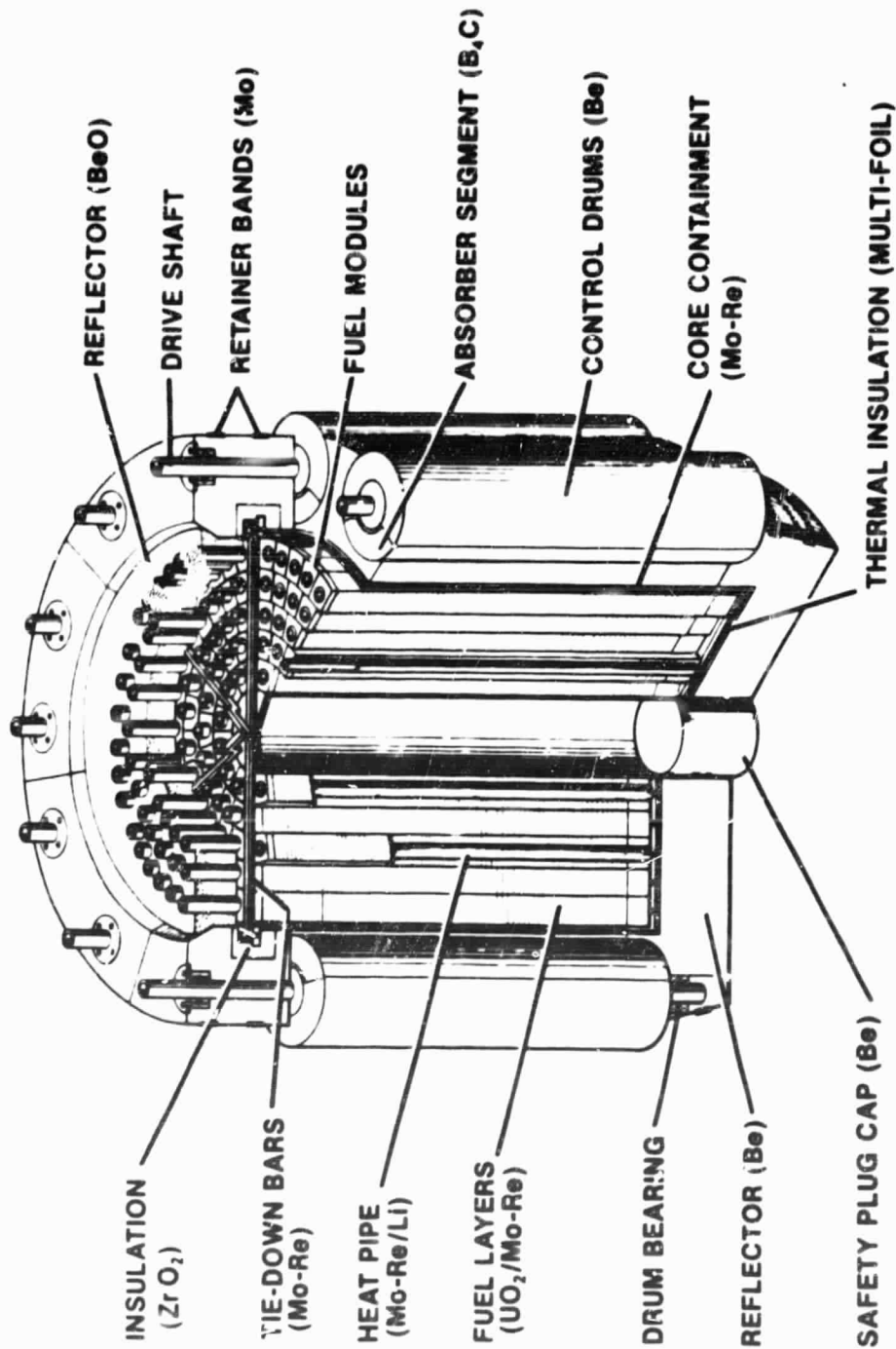
FIGURE 1 Cutaway view of the SP-100 baseline reactor design. This reactor contains 120 fuel modules built around Mo-13Re heat pipes with lithium working fluid, which remove the core heat at a temperature of 1500 K.

THERMAL SHIELD

HEAT PIPES

THERMOELECTRIC
PANELS
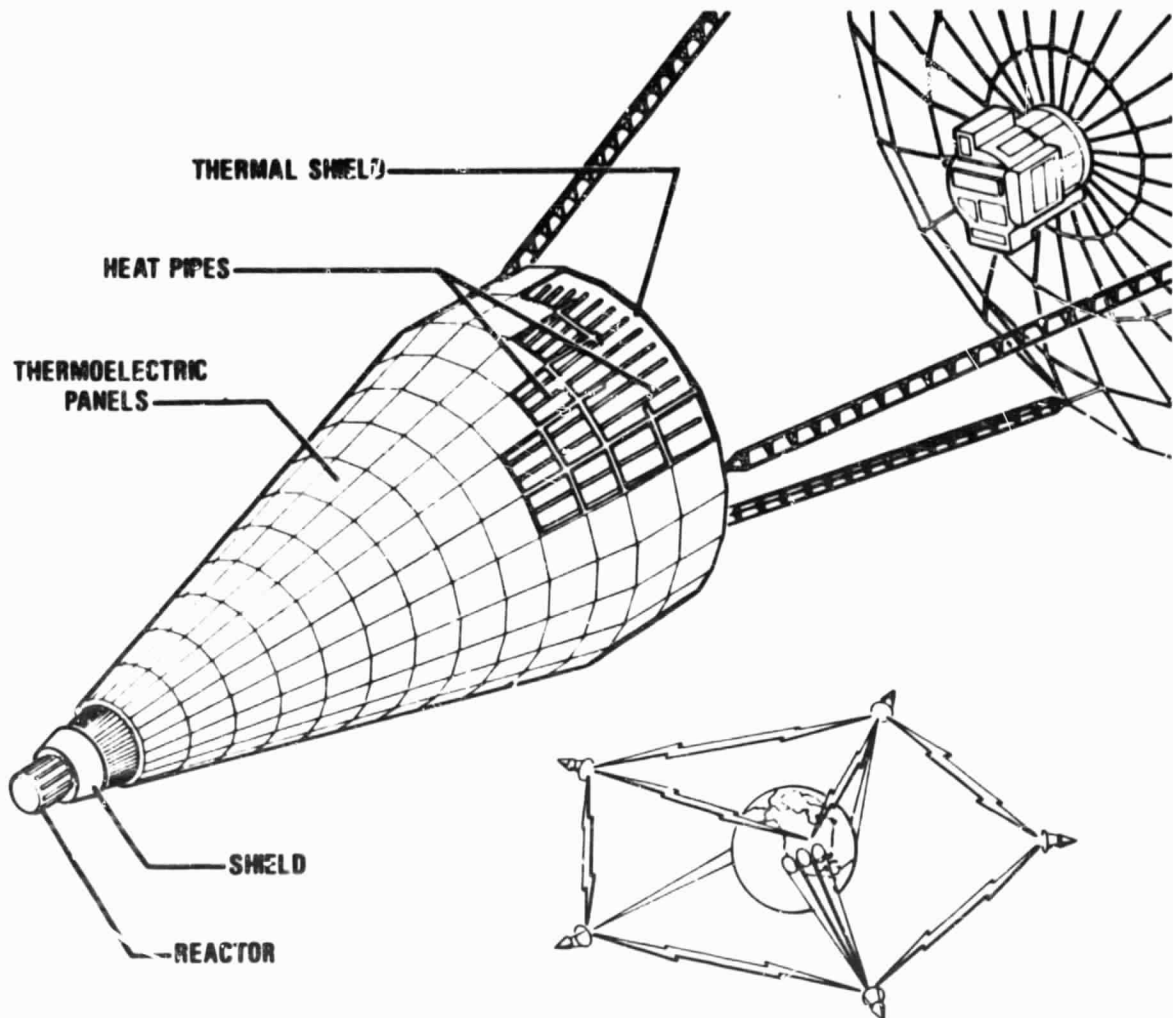
SHIELD

REACTOR

FIGURE 2 View of the SP-100 nuclear reactor space power system
showing reactor core heat pipes traversing the shield and forming a
truncated conical configuration that radiates reactor heat to an
encompassing array of thermoelectric converter panels. Reject heat is
radiated to space from the outer surface of this array.
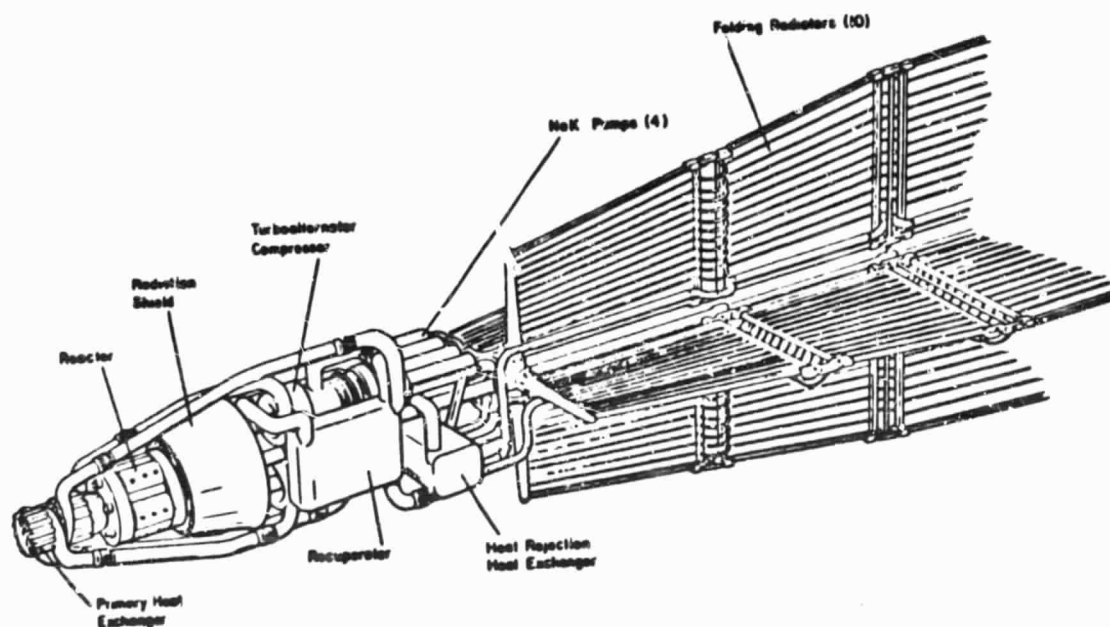
FIGURE 3 Brayton-cycle conversion system tied to heat pipe reactor.
This design features a two-fold redundancy made possible by a dual
heat exchanger that supplies heat from the reactor heat pipe bank to
two completely independent conversion units.
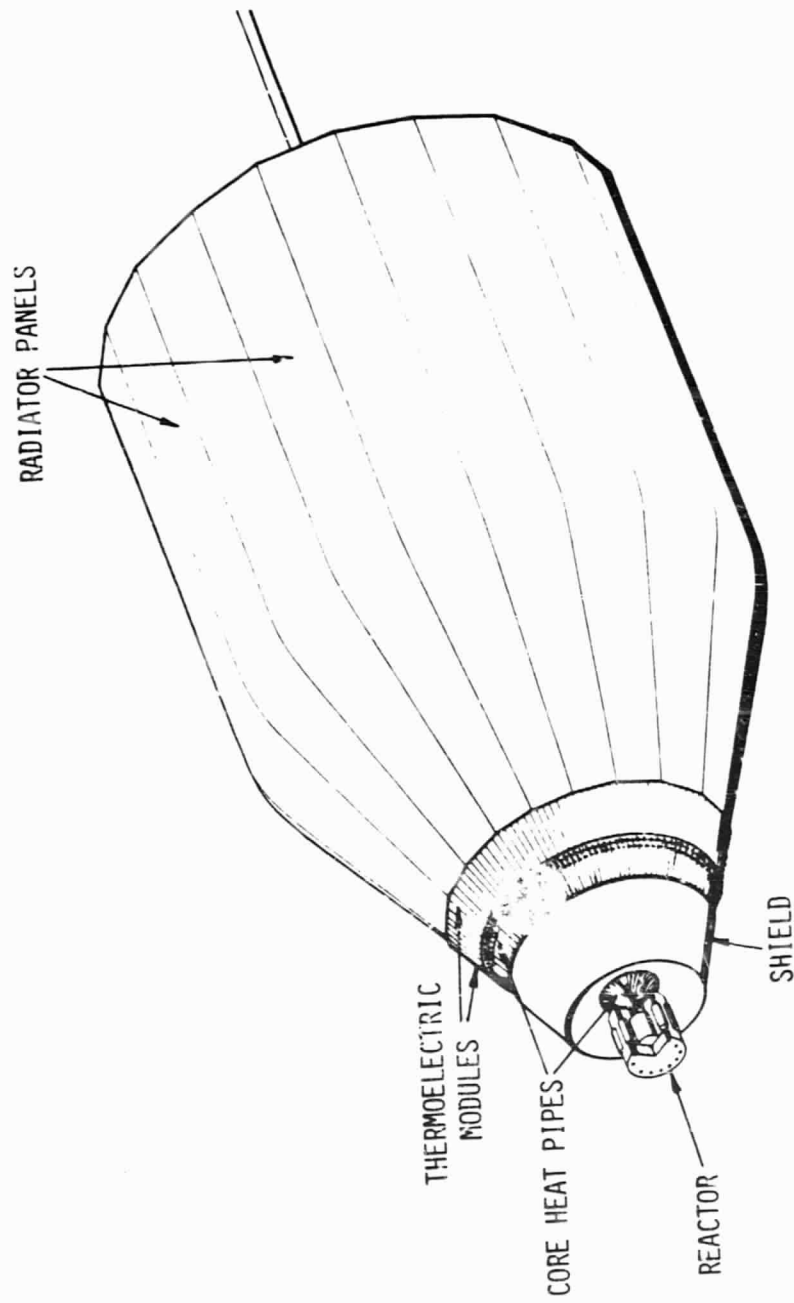
FIGURE 4 The 100-kW(e) predecessor of the SP-100 space power system. This system used lightweight titanium/potassium heat pipe panels to form the waste heat radiator surface designed to function at 775 K.

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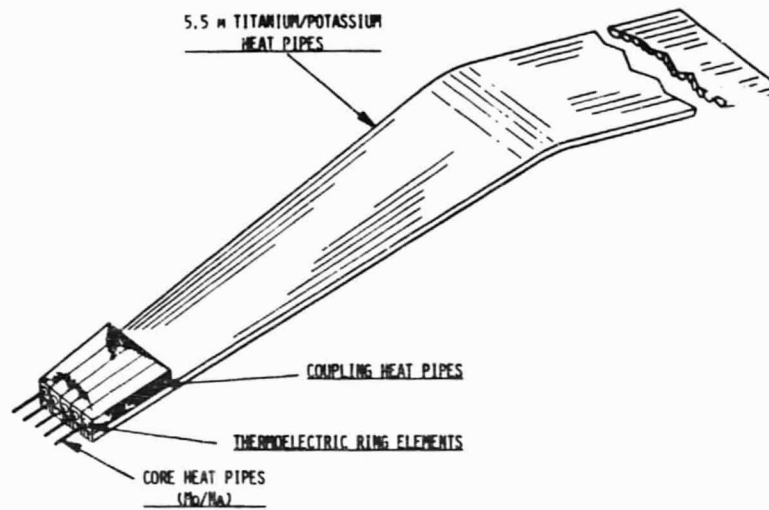
FIGURE 5 Radiator panel component of the space power system shown in
Figure 4. Titanium/potassium heat pipes, 5.5 m in length, are
connected in a redundant manner to compact thermoelectric conversion
modules.
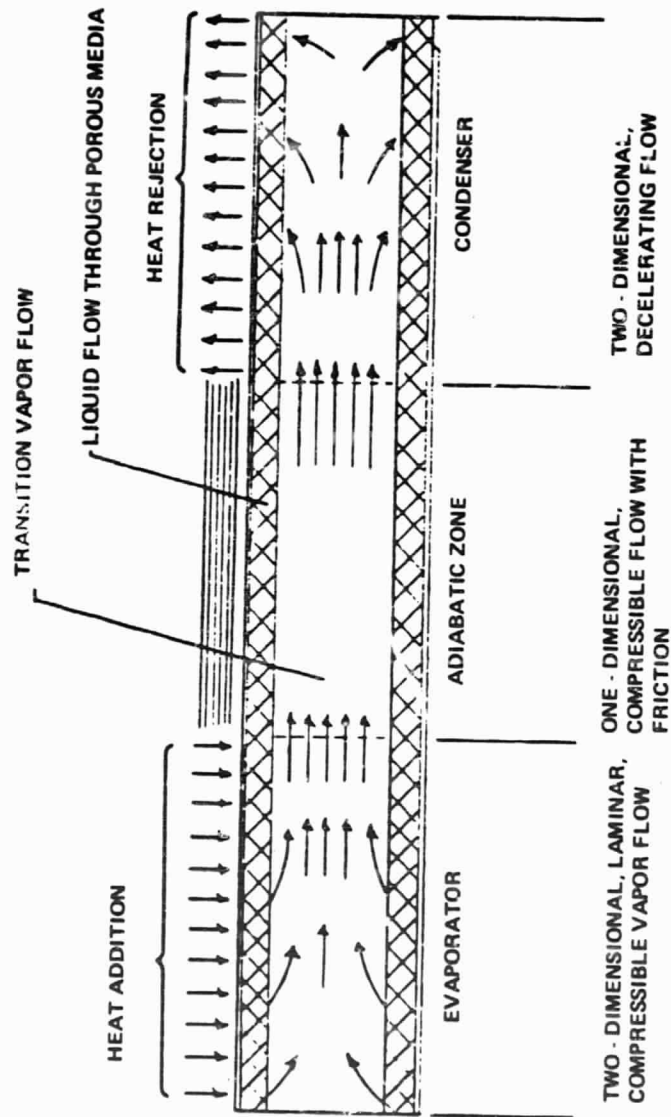
FIGURE 6   Heat pipe flow regimes modeled by the computer program HTPIPE.
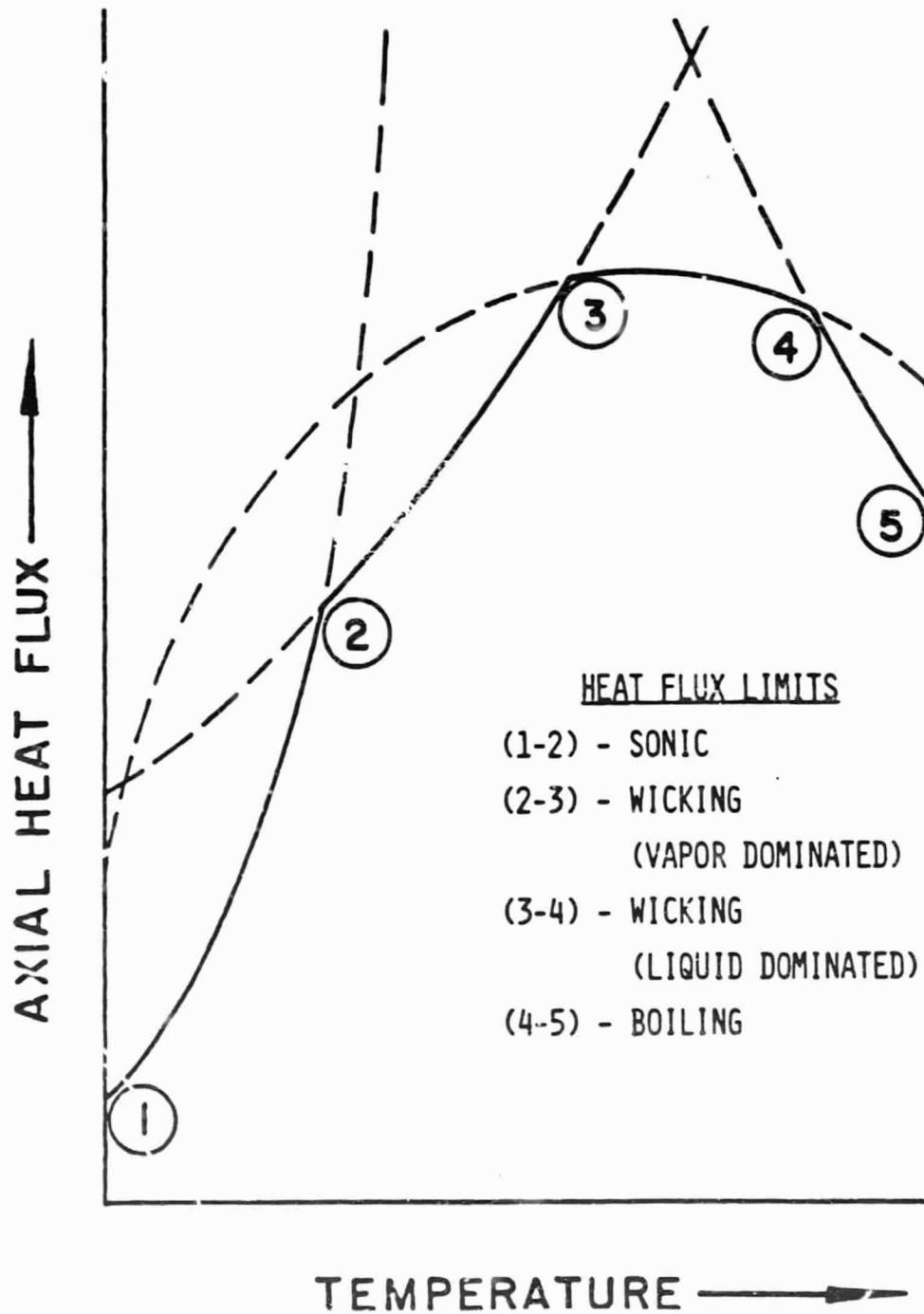
FIGURE 7 Phenomena limiting heat pipe performance. For lithium heat pipes the liquid-dominated wicking limit and the boiling limit are rarely a factor.
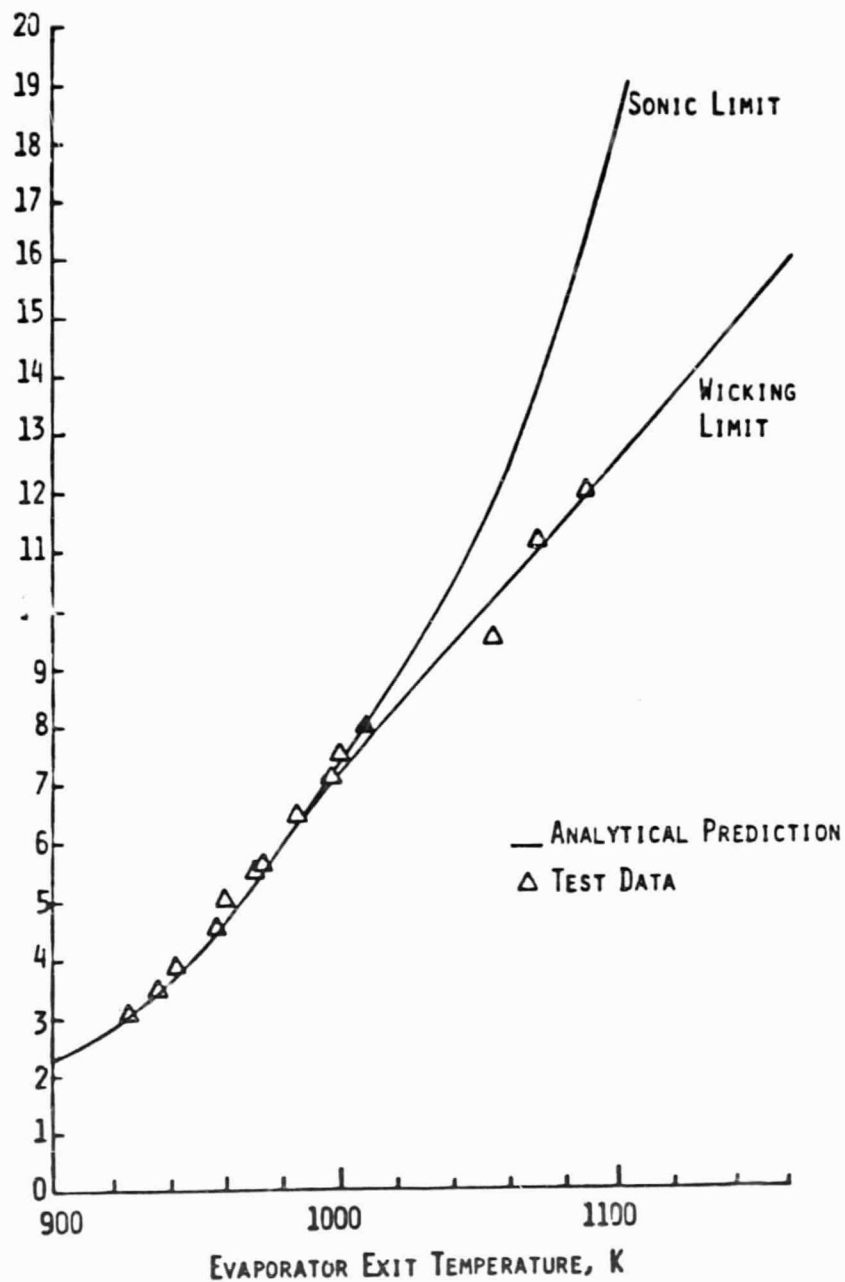
FIGURE 8   Example of benchmark experiment verifying the
performance-predictive capability of HTPIPE. The solid lines are
calculated for sodium in an annular wick with a capillary pore
diameter of 24 μm. The vapor space diameter is 14.1 mm, and the
evaporator, adiabatic, and condenser sections are each 0.60 m long.
Turbulent flow in the adiabatic section is assumed.
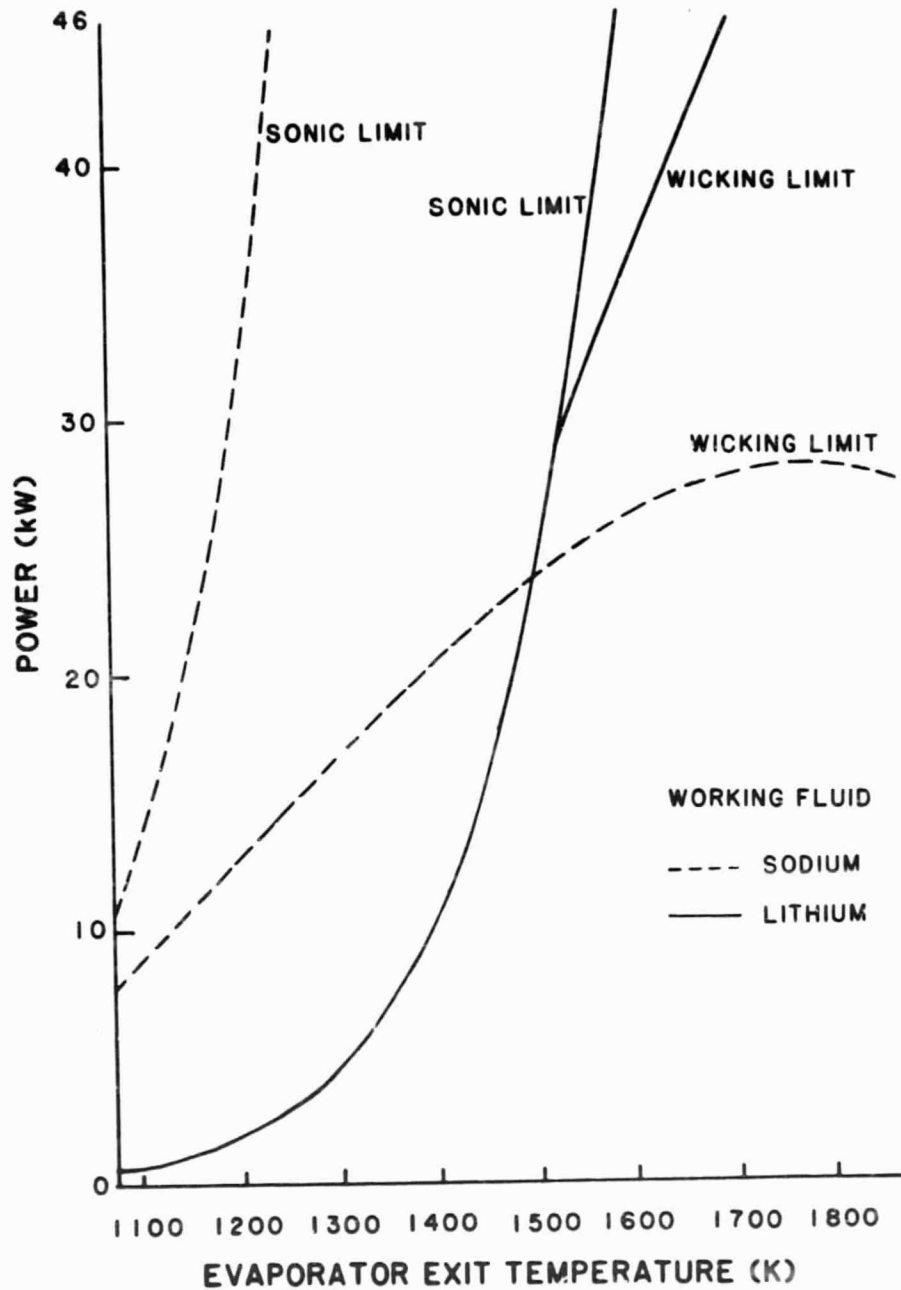
FIGURE 9 Comparison of heat pipe performance of lithium and sodium working fluids for an annular wick pipe with 14.1-mm inside diameter, 33-μm capillary pore diameter, two 90° bends with 0.2-m radius, and evaporator, adiabatic, and condenser section lengths of 0.30, 1.03, and 0.6 m, respectively.
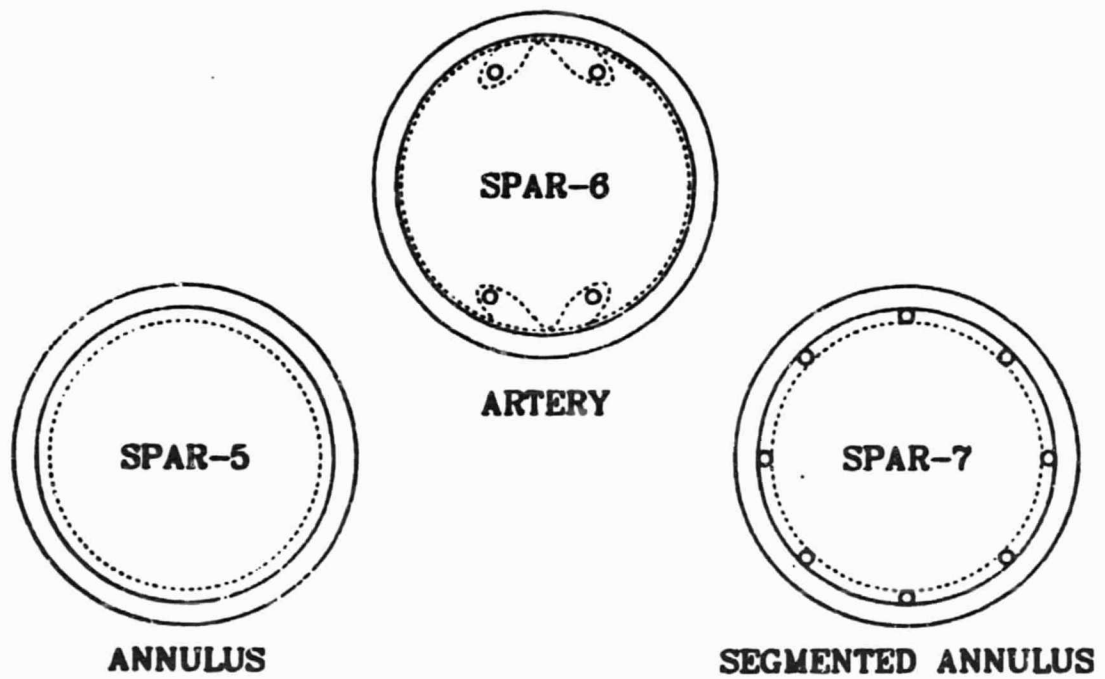
FIGURE 10 Composite wick configurations used in recent test heat pipes.
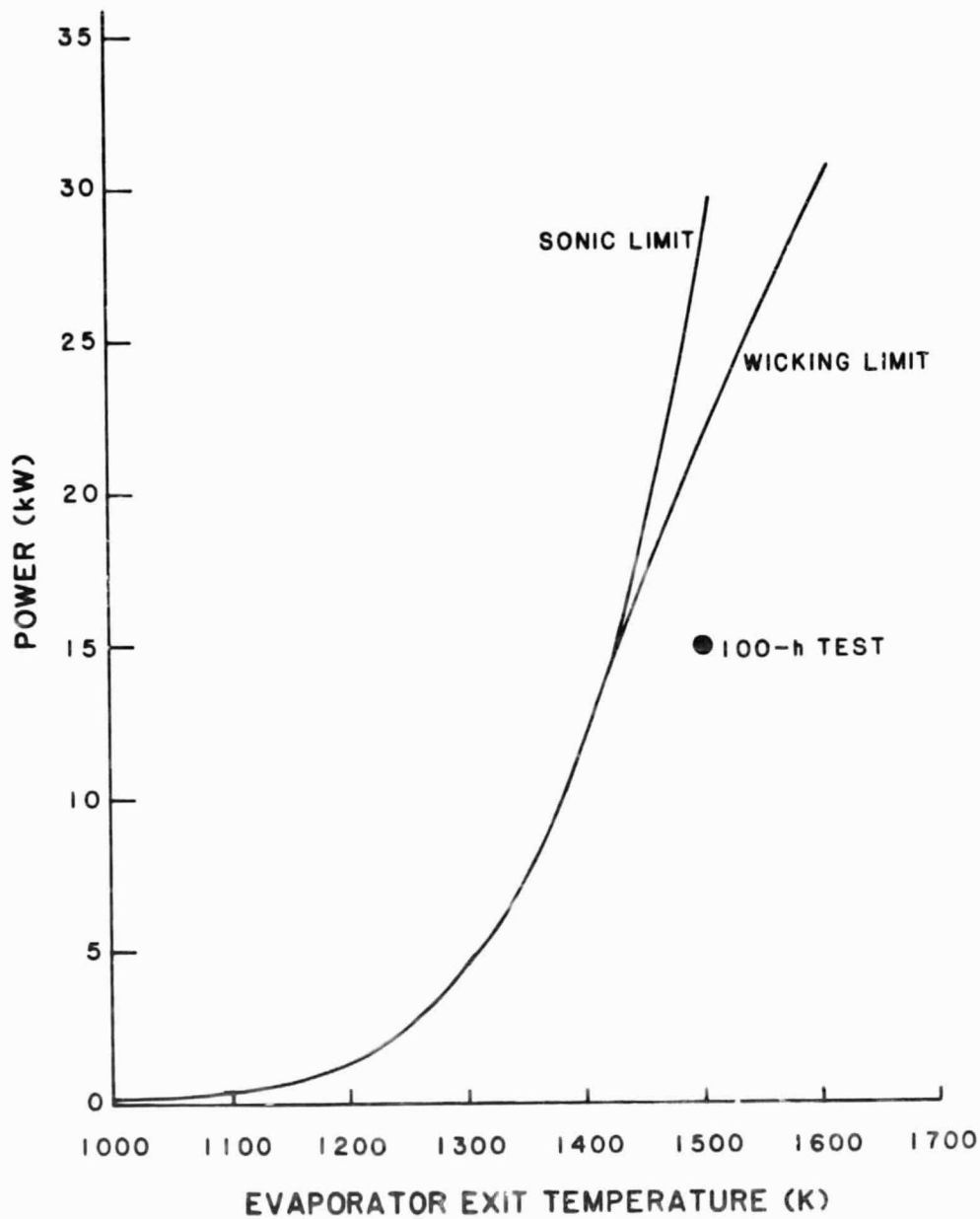
FIGURE 11 Predicted performance of SPAR-6 butterfly artery test heat pipe showing relation of 100-hour operating point to performance limits. SPAR-6 is a lithium heat pipe with an outside diameter of 15.9 mm, 45-μm capillary pore diameter, and evaporator, adiabatic, and condenser region lengths of 0.3, 1.03, and 0.6 m, respectively.
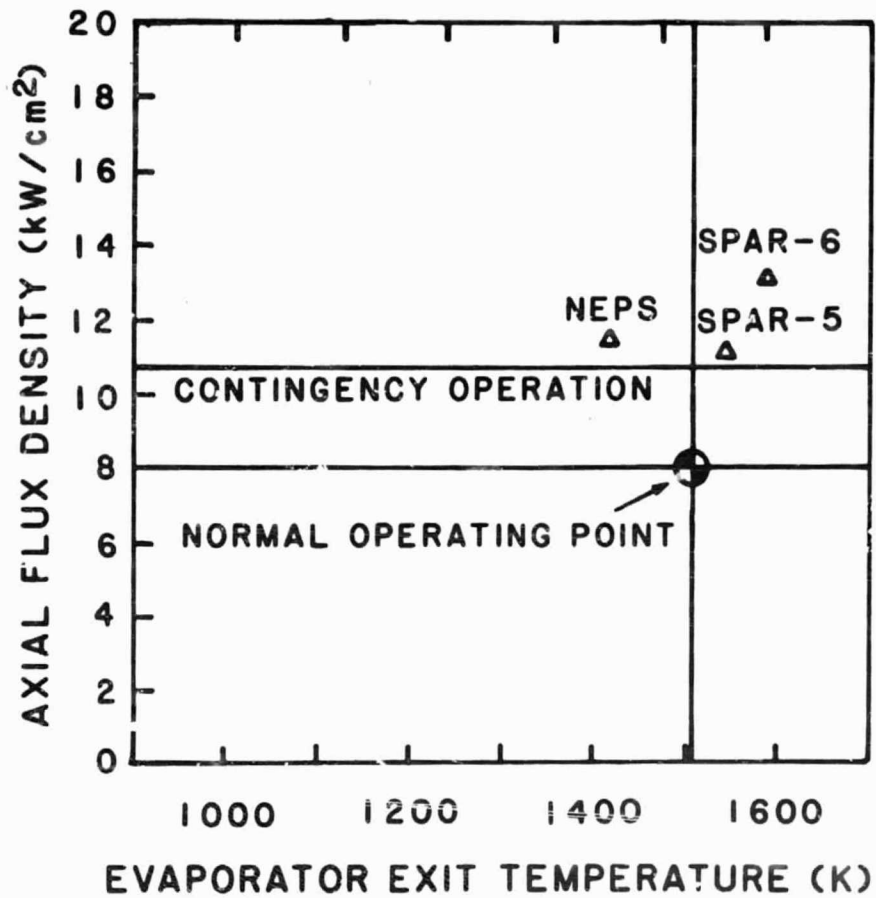
FIGURE 12 Axial heat flux achieved by several 2-m-long lithium heat pipes compared to the normal and contingency operating requirements for the 8-m SP-100 reactor heat pipes.
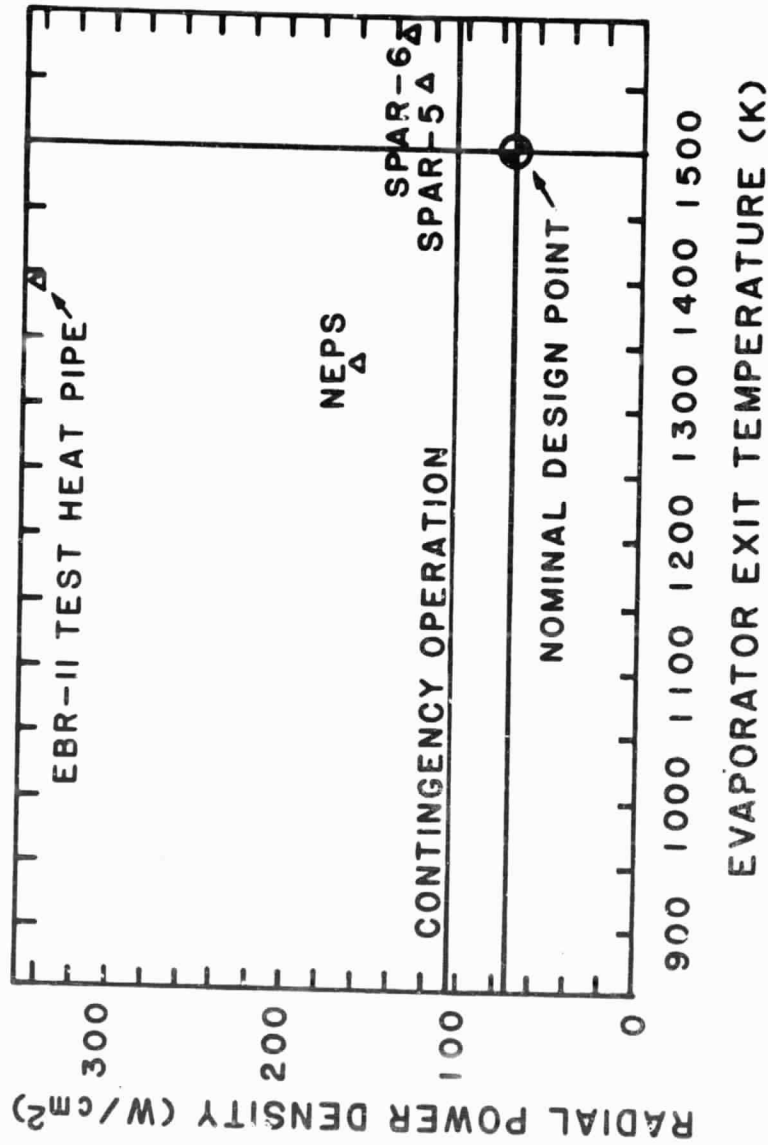
FIGURE 13 Radial heat flux values achieved for various lithium heat pipes compared to normal and contingency operating requirements for the SP-100 reactor heat pipes.
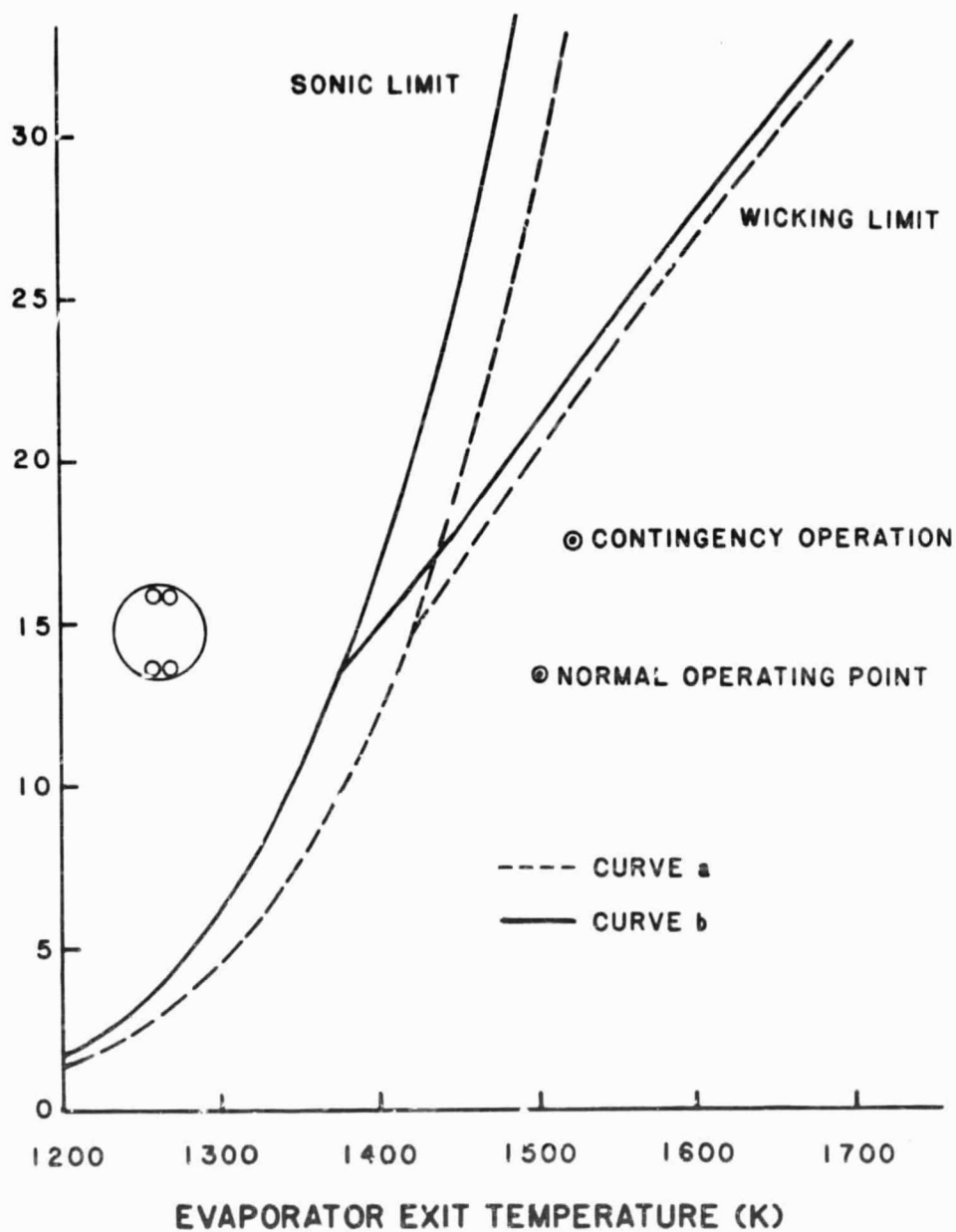
FIGURE 14 Performance predictions for 8-m-long heat pipes. Curve a is for 15.9-mm-diameter units with 30-μm capillary pore diameter, and curve b is for 17.5-mm-diameter heat pipes with 45-μm capillary pore diameter. Normal and contingency operating points are shown for the SP-100 reactor system design.
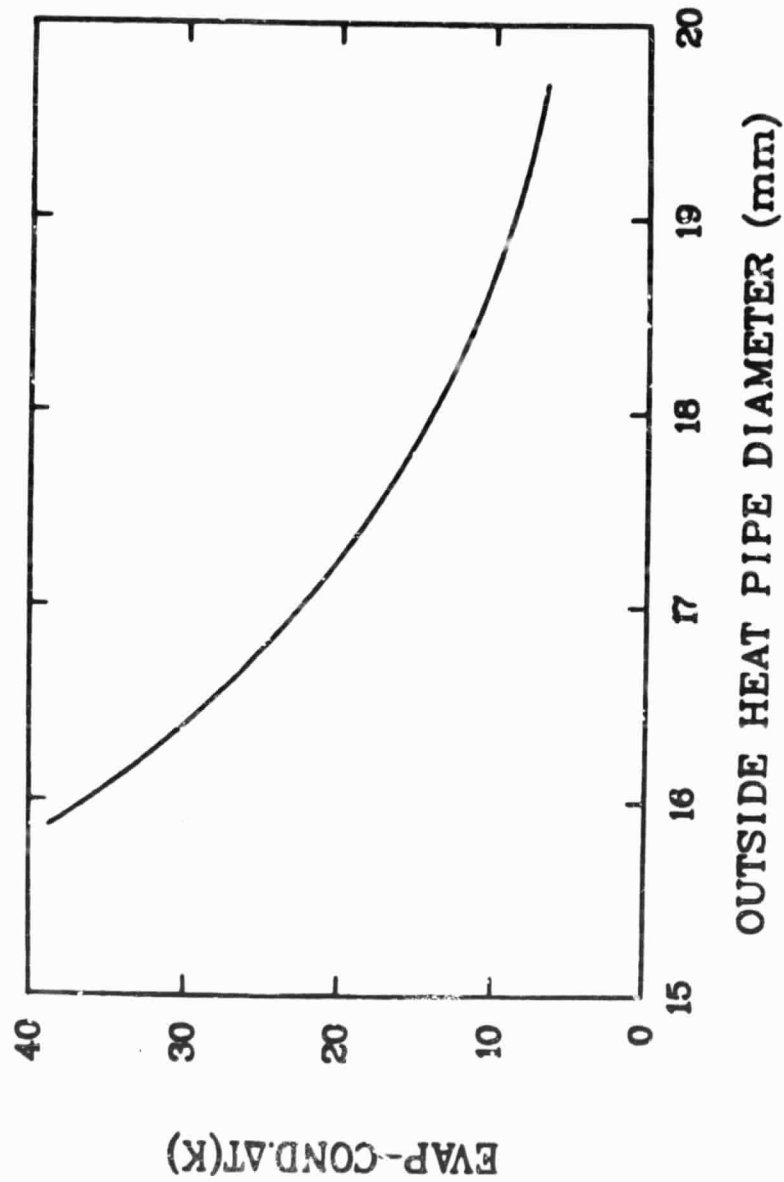
FIGURE 15 Predicted temperature drop versus heat pipe diameter for 8-m-long lithium heat pipes carrying 15 kW at 1500 K. The wall thickness is 0.89mm, and the capillary pore diameter is 30 μm.

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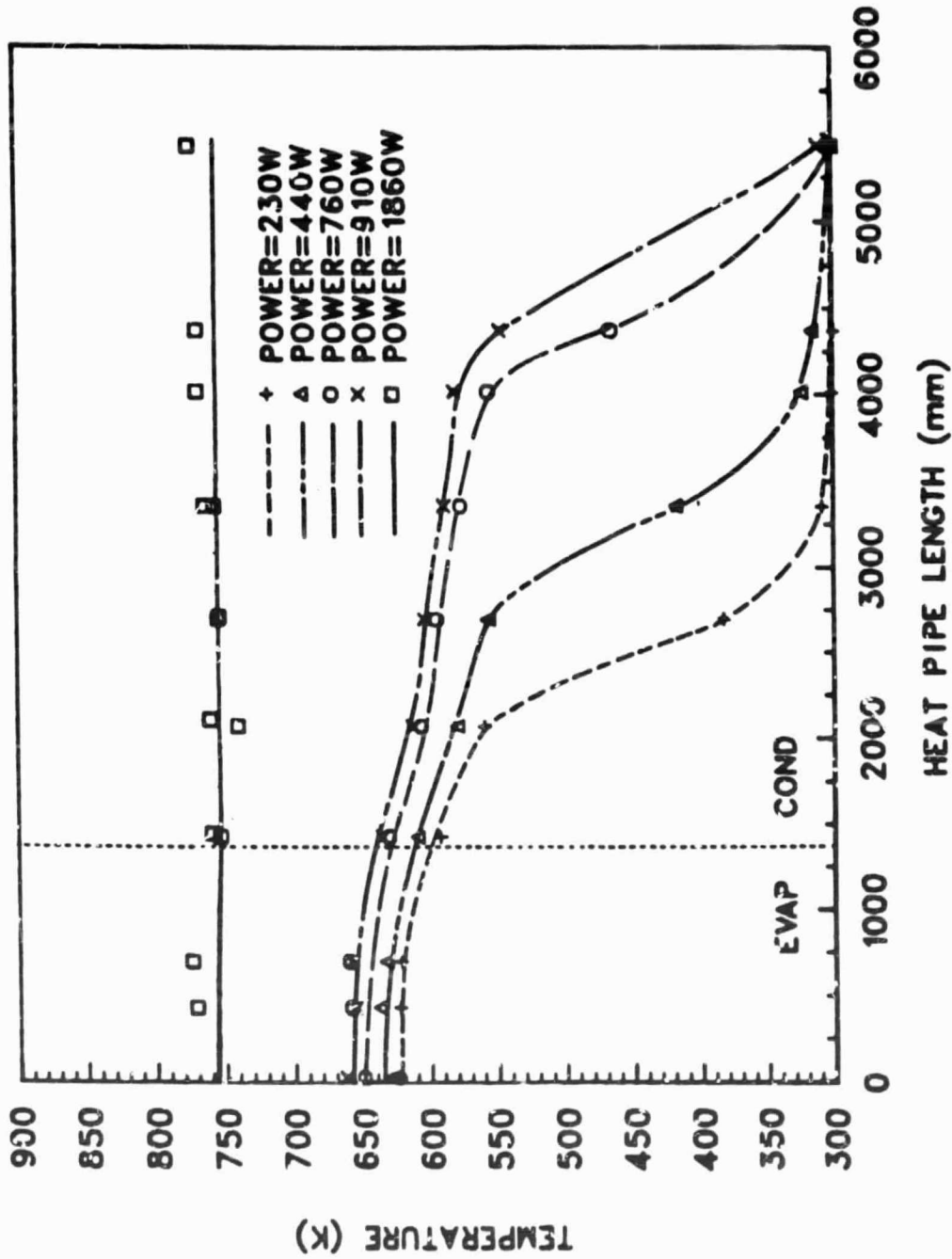
FIGURE 16 Temperature profiles obtained for a 5-m long titanium/potassium heat pipe for full-power operation and for fractional-power operation. Testing of this heat pipe confirmed the capability of starting up very long artery heat pipes with frozen working fluid and demonstrated operational stability when only part of the heat pipe was operational and the rest contained working fluid in the solid phase.
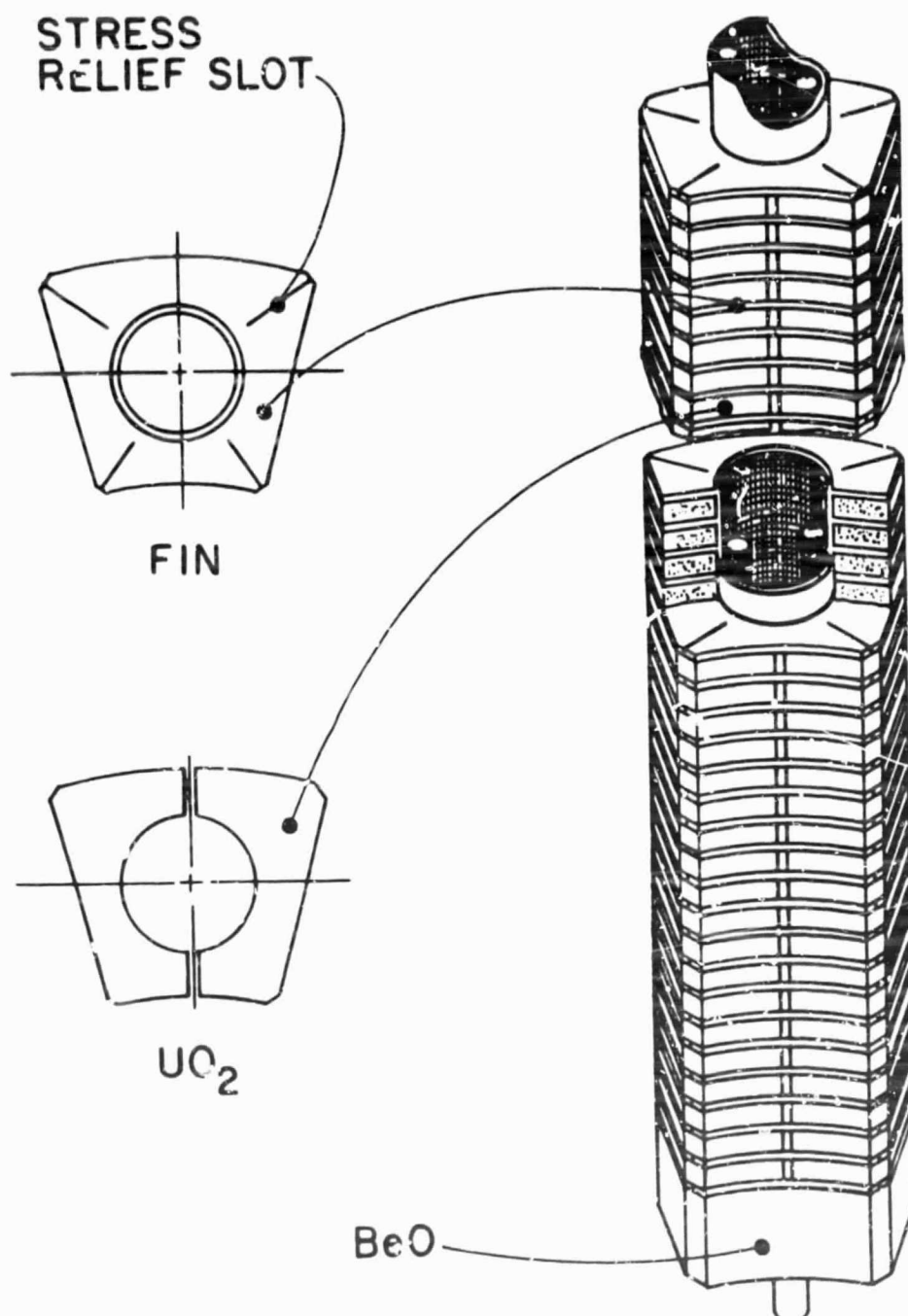
FIGURE 17 Fuel module configuration for the SP-100 baseline reactor design. The UO_2 fuel wafers are tightly sandwiched between Mo-13Re fins that are attached to the central lithium heat pipe. Because the heat flow is radially inward, the corners of the fuel wafers will be at the highest temperature—approximately 1700 K for normal operation.

REFRACTORY METALS FOR NUCLEAR SPACE POWER AND PROPULSION

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INTRODUCTION

During a 20-year period, from 1954 to 1974, a massive effort was organized and guided by the Department of Defense (DOD), the National Aeronautics and Space Administration (NASA), and the Atomic Energy Commission (AEC) to develop a refractory metals and alloy industry that could support materials needs for the rapidly developing nuclear and aerospace industries. These materials, which are based on the elements columbium (Cb), tantalum (Ta), molybdenum (Mo), and tungsten (W), are characterized by high melting points (refractoriness) and the retention of high strength at temperatures from 900° to 1799°C. Comparative properties are summarized in Table 1 (Perkins, 1961).

Refractory metals have been used since the early 1900s for specialty applications in chemical process and electronic industries, but it is only in the last 20 years that high-strength alloys have been produced as a full line of mill products on a tonnage basis. At the peak of this era, 11 major producers were manufacturing over 20 different alloys as bar, rod, plate, sheet, foil, tube, and wire (Tietz and Perkins, 1964). High-quality mill products were extruded, forged, rolled, and drawn routinely, with sheet sizes up to 40 in. wide by 120 in. long for some alloys of Cr and Ta. The spectrum and quality of products were comparable to those of stainless steels and superalloys. Much of this can be attributed to a major effort initiated by the Bureau of Naval Weapons and ultimately supported by DOD, the Air Force Materials Laboratory (AFML), NASA, and AEC to develop large-scale-production capabilities and practices for the manufacture of Mo, W, Cb, and Ta alloy sheet (Materials Advisory Board (MAB), 1966). Other programs were conducted in virtually all areas of consolidation and fabrication, including melting, extrusion, forging, rolling, and tube drawing and foil rolling. Advanced facilities were constructed, including fully integrated processing plants and an inert fabrication plant (Infab) in which refractory metals could be forged and rolled to 24-in.-wide sheet in an argon atmosphere to reduce surface contamination with oxygen and nitrogen (MAB, 1966).

TABLE 1 Properties of Refractory Metal Sheet Alloys

Characteristics	Cb Alloys	Mo Alloys	Ta Alloys	W
General Properties				
Melting point (°F)	4400-4600	4730	5425-5520	6170
Maximum service temperature (°F)	3000	3500	4000	4500
Physical Properties				
Density (lb/in.³)	0.29-0.38	0.37	0.60-0.61	0.70
Linear expansion (in./in./°F)	4.0-5.0	2.5-4.0	3.5-5.0	2.5-4.0
Thermal conductivity	30-50	50-70	40-50	60-70
Elastic modulus (psi x 10⁶)				
70°F	12-17	47	27-30	57
2000°F	10-15	28	16-20	48
3000°F	<10	<15	<15	30
Mechanical Properties				
Tensile strength (KSI)				
70°F	60-110	100-175	60-160	100-150
2000°F	15-70	20-90	20-60	30-90
3000°F	<5	5-15	5-30	15-30
4000°F	<2	<2	2-10	5-10
Elongation (%)				
70°F	10-30	5-30	10-30	<1
2000°F	10-70	10-30	10-50	20-40
Stress for 1% elongation (in./min x (KSI))				
3000°F	—	4-7	3-5	—
Fabricability Properties				
Recrystallization range (°F)	1800-2600	1800-2800	2200-2800	2400-3000
Forming temperature (°F)	50-500	50-1000	50-100	500-2500
70°F Bend ductility				
Base metal	0-2T	0-2T	0-2T	nil
Weld metal	0-4T	1T to nil	0-2T	nil
45° Brittleness	no	yes	no	yes
Lamination tendency	no	yes	no	yes

During the same time period (1954-1974), a very large supporting technology base for research, development, testing, and evaluation of refractory metal alloys was established in both industry and government laboratories. New alloys with improved strength and ductility were designed, tested, and qualified for use. The total scope of this effort was never estimated, but it was vast by any basis of measurement. As one example, NASA-Lewis Research Center (NASA-LRC) alone, from 1963 to 1974, contracted over $18 million in R&D to develop and evaluate alloys of Mo, W, Ta, Cb, and Cr (Klopp, 1974). The total professional staff at NASA-LRC ranged from 21 to 32 full-time investigators in the field of metallurgy and corrosion of refractory metals. A total of 215 staff and contractor reports on refractory metals were prepared under this program as indicated below:

Subject	No. Reports
W, Mo alloys	39
Cr alloys	24
Cb, Ta alloys	25
Surface coatings	20
Liquid-metal corrosion	33
Nuclear technology	66
Porous ionizer	8
	215

A summary of corresponding efforts by other NASA, DOD, and AEC laboratories during this period is not available (to the author's knowledge).

By the early 1970s, a superb technical base for supporting a major use of refractory metals in nuclear and aerospace applications had been established. Unique test facilities required to obtain data on materials properties under controlled environmental conditions, including ultrahigh vacuum (10^{-9} Torr), were constructed, and a large staff of trained personnel was active throughout the country in developing needed engineering design data. One unique high-temperature test facility at TRW Cleveland had 18 test stands with 10^{-9} Torr vacuum capability for controlled environment creep tests (Klopp, 1974). NASA-LRC had a fully integrated facility for melting, fabrication, and controlled environmental testing of refractory alloys. Other government laboratories of NASA, as well as the Air Force, Army, Navy, and AEC and industrial laboratories of both producers and users, had extensive research, development, and testing facilities. Vast quantities of data on alloy properties and behavior were being generated.

Two decisions in the early 1970s drastically altered the refractory metals industry and supporting technology base. Work on nuclear space power systems was terminated in January 1973 for an indefinite period. At about the same time, reusable surface insulation (RSI) was selected over coated refractory metals as the basic thermal protection

system for the Space Shuttle. Tne only continuing work was on the use of refractory metals in missile and spacecraft propulsion systems, jet engines, and a variety of industrial applications. This drastically curtailed activities at all levels of production, R&D, and testing/evaluation in both industry and government facilities. At NASA-LRC, for example, the professional staff specializing in metallurgy and corrosion of refractory metals dropped from 16 in 1972 to 1 in 1974, and the budget for refractory metal R&D decreased from $1.3 million in 1972 to zero in 1972 (Klopp, 1974). Similar decreases occurred throughout the industry.

In the intervening period since tnen, major changes and adjustments have occurred in the industry. Four of the major producers (Dupont, Universal-Cyclops, Stauffer Metals, and Westinghouse) ceased operations with refractory metals and either dismantled facilities or committed them to other uses. Production was curtailed by the remaining seven and was limited to the manufacture of those materials being used for ongoing applications in missile and spacecraft propulsion, nuclear programs, and commercial uses by the chemical and electronics industries. Much of the production equipment was dedicated to other uses or sold as the industry readjusted to a reduced volume market.

Of greater importance to the overall industry was the near-total collapse of the supporting technology base. Laboratories were closed, and equipment was dismantled and often sold. For example, the 18 high-vacuum creep test stands for environmental testing were returned to NASA. They were sold or donated, and their current status is unknown. Technical personnel were reassigned or took other jobs in different areas in nearly all industry and government laboratories that were involved in R&D or testing and evaluation of refractory metals. Much of the work that had been started was left unfinished. Data that had been gathered were filed away in numerous internal and government reports.

In 1974, a NASA study was unable to identify any near-term missions that would require refractory metals technology beyond the then state of the art (Siegel, 1974). This situation has remained largely unchanged until present. Now, advanced, compact nuclear systems are being reconsidered as portable thermal and electrical power sources for possible aerospace propulsion or other uses. Such systems will require the extensive use of refractory metals, some of which may be state of the art and some of which may be advanced materials. This paper reviews the current state of the art of refractory metal technology and assesses the problems that may be encountered in supporting a new technology in advanced space power systems. Critical problem areas that may need to be solved for adequate materials support are identified.

INDUSTRY STATUS

Alloy Production

The major producers of high-purity-vacuum (low interstitial) arc- or electron-beam-melted (EB-melted) refractory metals and alloys and the mill products currently in production are listed in Tables 2 and 3. Tungsten and its producers are not listed since it is no longer manufactured via an arc-melting route. All current commercial production of tungsten is by powder metal processes. It should be noted that all of the alloys listed in Table 3 also can be produced via a powder metal route and several suppliers of Mo, Cb, and Ta alloys as either powder metal or vacuum-melted and wrought materials.

Alloys for nuclear power applications involving lithium or potassium heat exchange media were produced in the past as high-purity arc- or EB-melted materials. High oxygen contents associated with powder metal products contaminated coolants and accelerated corrosion. Problems with producing ductile, crack- and pore-free welds with powder-metal-processed alloys were also encountered. In recent years, there has been a trend in the industry toward greater use of powder metal consolidation, particularly for the manufacture of near net shapes for cost reduction. The quality of powder metal Cb-base alloys has been improved significantly, and oxygen contents are now within the specification range for EB-melted and wrought alloys. Mechanical properties of powder metal and EB-melted wrought alloys also are comparable, and weldability appears to be good (Wadsworth et al., 1982a). These materials have not been evaluated for nuclear power plant use, however, and would have to be qualified before any large-scale use could be considered. Powder metallurgy Mo products in general are similar to those produced for the past 30 years. These alloys tend to present more problems with brittle fracture than arc-cast alloys, and the latter are still preferred for structural applications in aerospace systems. AMAX produces two grades of arc-cast unalloyed Mo: low carbon (less than 50 ppm) and normal carbon (more than 50 ppm). The low-carbon grade presents the least problems with brittle fracture and is preferred for aerospace structural applications.

The alloys listed in Table 2 are those currently manufactured on a routine, continuing basis for both aerospace and commercial applications. They are produced as a full range of mill products, although activities in tubular products are largely limited to a few Cb- and Ta-base alloys. Mo and TZM tubing are not being produced on a continuing basis but are available from several tube mills as specialty products. The standard sheet product for all materials currently is 24-in. x 48-in. maximum. The industry does produce some wider sheet at present but would need to use steel mill equipment for processing if wider products were needed on a large-scale basis.

Of the above alloys, only the unalloyed metals, TZM-Mo, Cb-1Zr, and the Ta-base alloys are maintained in open but limited stock. All

TABLE 2 Major Producers

Major Producers	Primary Mill Products
AMAX Specialty Metals Co., Parsippany, N.J.	Mo alloys
Wah Chang Corp., Albany, Oreg.	Cb, Ta alloys
Fansteel Metals, North Chicago, Ill.	Cb, Ta alloys
KBI Division Cabott Corp., Poyertown, Pa.	Cb, Ta alloys
National Research Corp., Boston, Mass.	Cb, Ta alloys

TABLE 3 Major Refractory Metal Mill Products

Alloy[a]	Bar	Plate	Sheet	Tube	Wire
Mo	X	X	X		X
TZM-Mo	X	X	X		X
Mo-30W	X	X	X		
Cb	X	X	X	X	X
Cb-1ZR	X	X	X	X	X
Cb-10Hf (C103)	X	X	X	X	
Cb-10W-2.5Zr (Cb 752)	X	X	X		
Cb-30Hf-9W	X	X	X		
Cb-33Ta-1Zr (F585)	X	X	X		
Cb-60Ta	X	X	X		
Ta	X	X	X	X	X
Ta-2-1/2W	X	X	X	X	X
Ta-10W	X	X	X	X	X
Ta-8W-2Hf (T-111)	X	X	X	X	X

[a]Arc- or EB-melted metal and alloy products.

other materials are produced on a batch basis as ordered and are not inventoried. Production of an alloy order can take as much as 16-20 weeks, depending on the product required and the level of activity in the industry. Tubing has the longest lead time.

The only alloys currently in production that are basic to nuclear uses and are fully qualified for use in nuclear power systems are Cb-1Zr and T-111. The Ta-base alloys that would be required for advanced nuclear power systems (T-111, ASTAR 811-C) are currently not being manufactured by most producers. T-111 is being made by National

Research Corporation on a routine production basis as a complete line of mill products, including welded tube. However, these alloys could be manufactured by all the major producers if needed, and a full capability for their production exists. In addition, Westinghouse, the developer and former manufacturer of T-111, has an integrated production facility in Ogden, Utah, that could be made available if needed to supply nuclear-grade Ta alloys. The plant currently produces zironium alloys but is capable of producing Ta alloys if needed. All producers expressed an interest and desire to supply the needs of any possible new applications by manufacturing virtually any of 30 different former commercial alloys. Producers are very receptive to toll production of new materials at the present time.

The refractory metal industry appears to be fully capable of supporting any major use of the three basic alloys that are likely to be needed for advanced space power systems: Cb-1Zr, T-111, and ASTAR 811-C. Melting and fabrication capacity for Mo, Cb, and Ta alloys is adequate, and supplies of raw materials are ample. Columbium, for example, is in a state of high excess capacity at present. The demand for mill products is down 30-45 percent from 1981, and there is an estimated excess production capacity of 5 million lb/yr (Marsh, 1982). The situation with tantalum is similar. Whereas a severe shortage existed a year ago, current demand for nearly all products is down, up to 50 percent in some cases. The production in 1982 will be 940,000 lb compared to 1,700,000 lb in 1980. Supply greatly exceeded demand, and prices are down. Tantalum powder, which was at a peak of $214/lb, is now available for $108/lb. Similar price reductions exist for Cb alloys, with Cb-1Zr redraw tube down from $118/lb last year to $72/lb today (Marsh, 1982).

The only major problem in the industry that could affect nuclear space power systems is related to the manufacture of very high strength alloys and to the production of seamless tubing, particularly Mo and Mo alloy tubing. The available extrusion capacity for large presses is limited to about 1500° F maximum billet temperature. This is adequate for low- to moderate-strength alloys but would be inadequate for higher strength materials. High-carbon Mo and W alloys such as TZC, WZC, Mo-Hf-C, and W-Hf-C are normally extruded from 2600° to 4000°F. The high-strength Mo-Hf-C alloys, for example, are extruded from 3500°F. Special heating equipment would have to be installed to process these materials on available presses. The largest extrusion press currently in use for Mo, Ta, and Cb alloy production is the 5,500-ton press operated by AMAX at Coldwater, Michigan.

Manufacture of seamless tubing is a unique problem that would need to be solved for nearly all materials except unalloyed Cb and Cb-1Zr. Tube production capability for refractory metals is limited and is particularly inadequate for Mo-base alloys. Large press and draw equipment must be built for efficient production of high-quality tube, and suitable tube drawing practices must be developed. AMAX estimates a 2- to 3-year lead time to develop suitable equipment and practices

to support large-scale use of Mo and its alloys as tubular products in
space power systems (Burman, 1982).

Testing and Evaluation

Refractory metals and alloys must be tested and evaluated under the
unique environmental conditions that either duplicate or simulate the
operating environment for a given application. This is particularly
true for long-time high-temperature exposure such as creep or
liquid-metal corrosion tests. The effect of atmosphere on creep
behavior is illustrated in Figure 1. Creep tests in a diffusion-
pumped vacuum chamber (about 10^{-5} Torr) give 2-3 times the creep
life for 1-2 percent strain compared with tests of the same material
under the same applied stress and temperature in an ion-pumped chamber
(about 10^{-9} Torr). This difference in purity level during test is
very large, and data from tests in moderate-vacuum equipment cannot be
used to design materials for use in a high-vacuum (i.e., space)
environment.

Although extensive ultralow-vacuum creep tests were conducted on
alloys such as T-111 or ASTAR 811-C in the late 1960s and early 1970s,
the results do not comprise an engineering data base that can be used
to design and qualify new systems today. New heats of these alloys
that will be produced for applications in the future will need to be
tested in depth for resistance to creep under ultralow-vacuum
conditions, since their behavior will not necessarily be the same as
that of previously tested materials. Testing may not need to be as
extensive, but each new heat of material will have to be qualified.
Alloys that have not been tested previously, such as Mo and its
alloys, will have to be creep tested in great depth under
ultralow-vacuum conditions. In addition, all tests must be of a
duration equal to or greater than the design life of structures. As
will be discussed, creep tests of refractory metals cannot be
accelerated, and short-time creep data cannot be extrapolated to
predict long-time deformation with any degree of reliability. A
design life of 10,000 hours will require a minimum of 10,000 hours of
creep data.

The industry at present is not capable of supporting creep testing
and qualification of either formerly used alloys (T-111) or new alloys
(Mo base) under ultralow-vacuum conditions. Most of the specialized
equipment built from 1960 to 1970 no longer exists (R. H. Titran,
personal communication). NASA-LRC has six operating ultrahigh-vacuum
test stands, and other laboratories such as Westinghouse and Oak Ridge
have modest capabilities. Any revived used of refractory metals in
nuclear space power systems will require extensive rebuilding of this
capability. It is estimated that a minimum of 12 units would be
needed to develop required creep design data for one alloy under
ultrahigh-vacuum conditions (R. H. Titran, personal communication).
Cost today probably would approach or exceed $100,000 per unit,

assuming the required ion pumps and basic related components are still available. It probably would take at least 2 years or more to construct the facility and an additional 1-2 years to conduct the tests. A similar facility would be needed for each additional alloy composition. This will be a pacing item in the design and development of advanced space power systems.

A parallel situation exists for evaluation of materials for resistance to attack by Li and K. Although an extensive data base exists for Cb-1Zr and T-111, current production materials would have to be requalified for behavior. Behavior is very sensitive to both alloy structure and composition, especially oxygen content, and it cannot be assumed that today's materials would behave the same as those of 15 or 20 years ago. New materials would require extensive testing. Such tests normally are conducted in closed-loop rigs that simulate closely service conditions (Stang et al. 1966). Test rigs in use 10 years ago no longer exist, and a new capability would need to be established.

No estimates of cost or lead time have been made, but this also is considered to be a pacing item. Long-term tests are necessary, and short-term data cannot be extrapolated for reliable life prediction.

Supporting Technology

The loss of a supporting technology base that has occurred over the past years is considered to be a major problem in the use of refractory metals for advanced nuclear space power systems. Technical personnel with the required knowledge, training, and experience in refractory metals for effective research, development, and engineering support are gone, and no one has been trained to take their place. Many are retired or near retirement and as such could comprise an important source of consultants to accelerate the transition for rebuilding a sound technical base. A new generation will have to be educated and trained in the unique and often complex characteristics and behavior of these materials.

Unlike stainless steels or superalloys, the refractory metals have not reached a design handbook stage. Their use is guided by a knowledge and understanding of behavior. Although vast amounts of data on alloy properties and behavior have been accumulated over the past 30 years, they have not been assembled into a useful engineering design data base. Many of the detailed data on properties and behavior are buried in numerous but scattered volumes of industry and government reports. Often only the highlights or a summary of selected parts of the work is published in the open literature. Many of these are no longer available from the contracting agencies but may be available from ASTIA or the National Aeronautics and Space Administration's Office of Scientific and Technical Information. Knowing which studies were conducted and which reports exist presents in itself a formidable task. This has been eased to some extent by

computerized data banks and search systems, provided that key words selected to catalog the work bear any resemblance to those selected for the search. Reports issued prior to the establishment of the National Technical Information Service, or the NASA Office of Scientific and Technical Information (in 1964 and 1962, respectively) are not in literature data banks and must be sought out by other means.

It would be very useful if most of the data accumulated over the past 30 years could be assembled as an engineering properties data bank on refractory metals. The task of educating and training new personnel as well as providing sound technical guidance for alloy testing, evaluation, and use would be greatly simplified. The lack of a sound engineering design data base will unduly increase costs and introduce delays as well as an element of risk in future work on the use of these materials, particularly in advanced nuclear power systems. The basis for such a data bank already exists in the form of the Mechanical Properties Data Center at Battelle Memorial Institute (BMI). However, data on refractory metals are not included in the BMI bank at present.

ALLOY TECHNOLOGY

General Behavior

Refractory metals and alloys are used primarily for their strength at temperatures above those where stainless steels or superalloys are used (i.e., above 800°C), or for their excellent resistance to corrosion by a wide range of chemicals and liquid metals. Excellent resistance to attack by liquid or gaseous lithium (Li), sodium (Na), and potassium (K) makes these materials prime candidates for heat exchangers and power trains in nuclear power systems. Each of the four refractory metals--Cb, Ta, W, Mo--and many of their alloys basically are resistant to attack and are compatible with the liquid-metal coolants and working fluids or vapors in advanced nuclear space power systems. The most important consideration with respect to liquid or gaseous-metal coolant corrosion is the oxygen content of both the coolant and the metal or alloy. High oxygen content in either case accelerates corrosion. The performance of a wide range of materials is considered to be adequate, and no need to develop new or improved materials has been identified. Detailed behavior of the various materials has been summarized by Stang et al. (1966) and will not be discussed further in this paper. Performance data are well documented, and a good technical data base exists. It would be helpful, however, if an engineering data bank accessible for computer search and containing all available data in one place would be established.

The most important mechanical properties of refractory metals and alloys are their strength at elevated temperatures and their toughness and ductility at low temperatures. The range of strengths available

in alloys that were considered for commercial production between 1954 and 1974 for each of the four refractory metal bases is summarized in Figure 2. Strength is expressed as the ratio of yield strength to density so that alloys can be compared on an equivalent weight basis. This is most important for aerospace applications, since weight is often a controlling factor in materials selection and design. As noted in Figure 1, Ta and W alloys have twice the weight of Mo and Cb alloys and as such would have to be more than twice as strong to justify their use on the basis of strength. The data in Figure 1 show that Ta alloys are not really strength competitive with Mo and Cb alloys below about 2800°F. Their use at lower temperatures, however, may be justified or required on the basis of toughness, ductility, and formability and/or weldability. Most of the high-strength Mo and Cb alloys present problems in these areas. The same is true with respect to tungsten. Although its strength-to-density ratio is superior those of other materials over most of the temperature range, its low ductility and toughness and limited weldability severely restrict its use as a structural material. The strength properties of each class of alloys will be presented in more detail in the following section.

Each of the refractory metals and its alloys undergoes a transition from ductile to brittle behavior with decreasing temperature, as summarized in Figure 3. Alloys of Cb and Ta have the lowest ductile to brittle transition temperature (DBTT), normally well below 0°C. The DBTT for Mo and its alloys is near room temperature, although it can be depressed below 0°C by controlled thermal-mechanical processing. Tungsten has the highest DBTT and presents the most difficulties in use. Problems with respect to the DBTT will be discussed in a following section. It is important to note that this is not a fixed temperature but that it can vary between wide extremes for all the refractory metals and alloys. The DBTT for any of these materials may be increased significantly by some or all of the following variables:

- o   Increased strain rate
- o   Interstitial contamination (O, C, N)
- o   Recrystallization
- o   Grain growth
- o   Intergranular precipitates
- o   Increased alloy content.

Detailed data on behavior of a wide variety of alloys are given by Tietz and Wilson (1965).

Another property of particular importance in aerospace applications is the modulus of elasticity. Comparative behavior of the refractory metals is summarized in Figure 4. Cb and its alloy have a low modulus at all temperatures and generally have to be used in heavier sections where stiffness or buckling is important. Tantalum alloys are stiffer, but when corrected for density, they are comparable to Cb.

Mo and W alloys have the highest moduli, and on a density-compensated basis, Mo and its alloys would be superior over a very wide range of temperatures.

A detailed summary of the behavior of each of these alloy systems has been assembled in book form by Tietz and Wilson (1965). The book presents comparative data on all aspects of mechanical behavior as well as a discussion of the various and often complex factors that influence mechanical behavior of refractory metals and their alloys.

Cb and Ta Alloys

Alloys of Cb and Ta have provided to date the basis for construction of compact nuclear space power systems. A summary of comparative properties for alloys that have been produced commercially at one time or another is given in Figure 5. The following two alloys have provided the basis for most of the space power systems designed and operated to date: Cb-1Zr and Ta-8W-2Hf (T-111). On a density-compensated-yield-strength basis, Ta-8W-2Hf would be superior at all temperatures from 2000° to 3500°F. ASTAR 811-C (TA-8W-1Re-1Hf), which has been taken to an advanced state of development for space power use, has similar properties but was not included in this summary figure (Figure 5). This material has improved creep properties compared with T-111, as shown in Figure 6. For a given time and temperature of exposure, ASTAR 811-C can operate at about twice the stress for 1 percent deformation by creep compared with T-111. For example, at 2000°F, the stress for 1 percent creep in 10,000 hours averages 15 KSI for T-111 and 30 KSI for ASTAR 811-C (Figure 6). The strongest Cb-base alloys (Cb-16W-20Ta-5Mo-2Zr-0.15C, Cb-132M and Cb-21W-1Zr, AS30) are comparable to T-111, as illustrated by the data in Figure 7. On a density-compensated basis they would be stronger than T-111 but not as strong as ASTAR 811-C. The Cb alloys, however, would present more problems in fabrication and welding, and they have not been fully characterized and qualified for use in space power systems. Extensive testing and evaluation would be required.

The commercially available Cb and Ta alloys such as Cb-1Zr, T-111, and ASTAR 811-C are considered to be wholly adequate for a wide range of nuclear space power applications. They have been tested and evaluated in depth and have been tested in operational units. These alloys would present the least problems and could be brought to an operational state of readiness in the shortest time at the least cost compared with all other materials. They are ductile, weldable, and fabricable and have DBTTs well below 0°C. In addition, they are very resistant to attack by liquid and gaseous Li, Na, and K.

Mo and W Alloys

Molybdenum and tungsten alloys are of particular interest where ultrahigh strength is required (turbines) or where resistance to

nuclear fuels is essential (heat pipes). Comparative properties of
the weakest and strongest alloys are shown in Figure 8. On a
density-compensated-strength basis, TZM-Mo is superior to tungsten at
all temperatures to 3600°F. Only a W-25Re alloy is higher in
strength above 2500°F. TZM-Mo is also stronger than the best Ta and
Cb alloys at temperatures to 2600°F (Ta) and 3500°F (Cb) on a
density-compensated basis. Creep properties of TZM-Mo are summarized
in Figure 9. The alloy is comparable to T-111, with a stress for 1
percent creep in 10,000 hours at 2000°F between 15 and 25 KSI. On a
density-compensated basis, however, it would be twice as strong as
T-111 and comparable to ASTAR 811-C. Above 2600°F, TZM-Mo
recrystallizes, and creep properties would not be as good.

The major problem in using either Mo or W and its alloys is their
high DBTT and their tendency to be brittle in the recrystallized state
(Figure 10). These materials are very notch sensitive and can behave
in a brittle manner if not properly prepared, designed, and used. In
addition, although Mo, W, and their alloys can be joined by welding,
the weld joints may be brittle. Embrittlement of both base materials
and welds will be a limiting factor in the use of these materials for
nuclear space power systems and represents a critical problem area in
which more work in needed. This is discussed in more detail in the
following section.

One approach to solving the embrittlement problem with Mo and W is
to alloy these metals with rhenium (Re). As shown in Figure 10, the
addition of 30-35 Re to Mo or W reduces the DBTT to below room
temperature. However, Re is a very scarce and expensive material, and
the industry at present could probably not support any major use as a
large alloy addition. Development of an adequate Re supply could be a
pacing factor, depending on the net total requirement for any new
application. In addition, it should be noted the Mo-Re and W-Re
alloys have had minimal testing and evaluation and little is known of
their behavior in terms of long-term creep, liquid-metal corrosion,
nuclear fuel compatibility, thermal stability, and DBTT behavior--to
mention just a few important properties. The properties and behavior
of these alloys have been summarized by Jaffe and Sims (1958) and
Lundberg (1981). Lundberg concludes that Mo-10-15 Re alloys offer an
advantage over pure Mo in space reactor core heat pipes with respect
to both DBTT and high-temperature strength. TZM-Mo, however, may be a
more practical alternative provided that welding problems can be
solved (brittleness).

Mo and TZM-Mo have been fairly well characterized with respect to
attack by liquid- and gaseous-metal coolants (Stang et al., 1966), and
minimal additional testing will be needed. The data base on long-time
and low-vacuum creep, however, is very weak, and extensive testing
would be required to develop essential design data for the wide range
of product forms. Unlike Cb and Ta alloys, Mo and TZM-Mo are
strengthened by cold work, and mechanical properties vary greatly with
deformation structures. Each product form has its own unique set of
properties, depending on the structure produced. Only fully

recrystallized materials of uniform grain size and shape will have similar properties for all product forms.

Mo and its alloys are not readily available in tubular form. Although tubing has been produced, it is not a routine commercial mill product. Major programs on manufacture of Mo and Mo alloy tubing will be needed to support any extensive use of this material in space power plants.

Coatings

If refractory metals and alloys are heated in air, they will oxidize at rapid rates. In addition to high rates of surface recession, oxygen and nitrogen diffuse into the alloy that with Cb and Ta alloys results in hardening and embrittlement. None of the commercial alloys developed during the past 30 years have any inherent resistance to oxidation. They must be protected by surface coatings of more oxidation-resistant materials during heating in air or oxidizing atmospheres at temperatures above about 600°C.

Four types of surface coatings have been used successfully to protect these materials (National Materials Advisory Board (NMAB), 19˙.): (1) silicides, (2) aluminides, (3) noble metals, and (4) Ni-Cr alloys. The most successful and widely used coatings are the disilicides of the base metal to which the coating is applied (i.e., $MoSi_2$ on Mo). Normally, coatings are 2-5 mm thick. Their useful life in air is summarized in Figure 11. Life tends to be governed by interdiffusion between the active coating elements and substrate elements and hence is an inverse exponential function of increasing temperature. Thinner coatings (1-2 mm) have a useful life near the lower side of the scatter band, while thicker coatings (4-5 mm) are near the upper side (Figure 11). As can be seen from Figure 11, life expectancy at temperatures above 2200°F is less than 1,000 hours. Temperatures of less than 2000° F are necessary for a useful life in excess of 10,000 hours. Repeated heating and cooling (thermal cycling) accelerates wear-out of silicide coatings.

Twenty years of intensive research has done little to improve performance, and the current state of the art probably is the best that can be achieved without a major technical breakthrough. Current coating process technology is geared largely to small components, although fabricated components up to 6 ft in diameter are being coated routinely. Coating process technology is not adequate for larger components, and major process innovations will be needed if coatings are required on large nuclear power system components. Coatings generally are not required for refractory metals heated in the vacuum conditions of space but may be needed for ground testing unless tests can be run in large vacuum or inert atmosphere chambers.

Contamination

Refractory metals will react with carbon and nitrogen as well as oxygen over a wide range of low partial pressures or activities of these oxidants. This can present major problems when the metals or alloys are heated in contaminated "inert" atmospheres or moderate vacuums (10^{-4}-10^{-5} Torr). The carbide and oxide phases stable for Cb and Mo heated at 2150°F are shown as a function of equilibrium oxygen pressure and carbon activity in Figure 12. Cb will not react with either carbon or oxygen if log a_C is less than or equal to -6.5 and log P_{O_2} is less than or equal to -21 atm. With equilibrium carbon activities above 10^{-6} or oxygen pressures above 10^{-21} atm, Cb can react to form carbides or oxides, depending on the P_{O_2}-a_C levels in the gas at 2150° F. The boundaries for Mo are at much higher levels, 10^{-1} for carbon and 10^{-12} for oxygen. Molybdenum and its alloys will have less tendency to react in impure atmospheres due to a lower stability of oxide and carbide phases.

These reactions and phase equilibria are of considerable importance for use of refractory metals in space nuclear power systems. Contamination levels at which reactions can be expected can be defined, and reaction kinetics can be studied under controlled conditions that simulate real environments. For example, Figure 13 illustrates the reaction of Cb with oxygen and carbon to form external oxide or carbide scales as a function of equilibrium oxygen pressure and carbon activities at 3000°F. Studies of such reactions show that rates of reaction are very low when CbC or CbO_2 can form as external scales and that protective coatings may be needed only under conditions where Cb_2O_5 can form. Kinetics of internal carburization and oxidation also appear to vary significantly with small changes in gas equilibra.

Few definitive studies of reactions of these alloys in atmospheres of low oxygen and nitrogen pressure or carbon activity have been made, and guidelines do not exist for using the alloys in impure atmospheres. Kinetic studies of internal contamination as a function of exposure conditions are particularly needed. Boundary conditions of time, temperature, and gas composition for use without detrimental internal contamination need to be defined.

Of equal importance is the fact that few definitive studies of the effect of internal contamination on mechanical behavior have been made. Although it is known that contamination may embrittle refractory metals, the precise effects of specific contaminants on various alloys have not been established with any degree of certainty. For example, Figure 14 shows the effect of oxygen and nitrogen contamination on the DBTT of welds in a high-strength Cb alloy (B-66). Typical of most data, they illustrate a relative effect but do not define property-composition relations to the extent that data can be used to control contamination through process or product specifications. Detailed studies of the individual and combined effects of C, O, and N at low levels of concentration (ppm range) on

mechanical behavior are needed for virtually all alloys. None of these alloys will be stable in even the best vacuum, and internal contamination is bound to occur during long-time exposure at high temperatures.

The effectiveness of currently available coatings to protect from internal contamination during exposure to vacuum and reactor atmospheres has not been studied. Coatings based on Si, Cr, and Al most likely will not be effective in vacuum, since the active coating elements have high vapor pressures. It may be necessary to develop new coatings based on noble metals or ceramics for effective protection of refractory metals from internal contamination in nuclear space power systems.

PACING PROBLEMS

There are a number of pacing problems that will need to be addressed early on in any program to develop advanced space power systems based on these materials. Several of these have already been discussed and are listed below as a summary:

o   Alloy production: (1) high-temperature (above 2500°F) extrusion capability and (2) seamless tube manufacture for high-strength alloys and Mo.
o   Testing and evaluation: (1) ultrahigh-vacuum creep test facility and (2) liquid- and gaseous-metal-corrosion facility (test loops).
o   Supporting technology: useable engineering data bank.
o   Alloy technology: (1) ultrahigh-vacuum creep data to 10,000 hours, (2) coating processes for large structures, and (3) coatings to protect from internal contamination in vacuum or reactor atmospheres.

There are three other pacing problem areas in which work should be initiated as soon as possible to support development of reliable space power systems: (1) alloy embrittlement, (2) weld embrittlement, and (3) creep life prediction. These are discussed in the remaining sections of this paper.

Alloy Embrittlement

Most of the technical problems encountered in the use of refractory metals are related to one form or another of embrittlement where normally ductile materials behave in a brittle manner. The previous discussions pointed out one type of this behavior, in which alloys can

be embrittled as a result of contamination by C, O, or N during use. In other cases, alloys are embrittled by a change in structure (recrystallization) or by welding. All of these phenomena are interrelated in that they most likely involved an interstitial contaminant--O, C, or N.

Recrystallization and weld embrittlement possibly are the most misunderstood aspects of alloy behavior, particularly for Mo and its alloys. It is common to hear Mo described as being inherently brittle in the recrystallized state and nonweldable. Neither statement is true: recrystallized Mo can be very ductile, and sound, ductile welds can be made with molybdenum.

A recent paper by Kumer and Eyre (1980) shows that brittleness in recrystallized molybdenum is the result of oxygen segregation at grain boundaries. Fracture stress is inversely proportional to the segregated oxygen level. Adding small amounts of carbon to Mo reduces the driving force for oxygen segregation and increases the tolerance for oxygen. If the grain boundary oxygen level can be reduced from 0.47 to 0.11 percent, fracture stress will increase to the point at which cracks propagate elastically and ductile behavior results. This work shows that Mo is not inherently brittle and holds forth the promise of controlling ductility through balanced composition.

Excess carbon in Mo also can have an embrittling effect, but by a different means. If Mo with 20 ppm C is heated to 2750°F for recrystallization, all of the carbon goes into solution and precipitates as carbides along grain boundaries during cooling (Figure 15). This precipitation cannot be suppressed, and carbon cannot be quenched into solid solution. During subsequent working, internal cracks are formed along carbide arrays, and the material may be brittle. However, if this material is recrystallized at 2150°F, less than 2 ppm C is soluble, and carbides do not form on the grain boundaries during cooling. Cracks are not formed internally during subsequent working, and the material remains ductile. The brittleness or ductility in this case is controlled by an in-process anneal between extrusion and rolling. The anneal must be adjusted to prevent dissolution of more than a few ppm C in order to prevent internal cracking during further working and subsequent brittleness. This is not standard industry practice at present.

Another example of ductility or brittleness in recrystallized Mo is given in Figure 16. Here a machined TZM-Mo tensile blank was wrapped in Ta-foil and recrystallized in vacuum (10^{-5} Torr) for 3.5 hours at 2550°F. Elongation at room temperature was 4 percent, indicating a DBTT of more than 20°C. However, when 1 mm of material was removed from the surface by etching, a similar sample had 32 percent elongation at 20°C. Repeated tests of this type revealed that recrystallization embrittlement was in fact the result of surface contamination with oxygen to a depth of 0.5-1 mm during the vacuum anneal. Removal of the contaminated surface yielded a high-ductility recrystallized structure for all samples heated up to and including 3000°F. This material is very ductile in the recrystallized state with a DBTT of -35° to -65°C in bending at 1-in./min load rates.

This is not to say that recrystallized TZM-Mo will be ductile at room temperature under all conditions if the contaminated surfaces are removed. The DBTT of Mo is a function of structure, composition, and strain rate. Detailed studies of the behavior of one grade of TZM-Mo indicate that for a particular composition and purity, the DBTT is an inverse linear function of hardness of the alloy for any given velocity of deformation. At any hardness level, the DBTT increases with increasing rate of bending. These relations are shown in Figure 17. The plots were used to derive an empirical relation that defines the DBTT in terms of material hardness at room temperature (H, in Rockwell A units) and velocity of loading at the test temperature (V, in in./min) as follows

$$DBTT \ (T - {}^{o}C) = 5.88V^{0.119}(67.4 - H) - 94$$

This relation has been found to be valid for wrought or recrystallized TZM-Mo sheet or bar at loading velocities of 1-10,000 in./min. The effect of variations in microstructure and grain size is incorporated in this equation only to the extent by which these factors influence hardness (strength) at room temperature.

This is a purely empirical relation, but one that is very useful for estimating the conditions under which TZM-Mo will be ductile or brittle by merely measuring the hardness in Rockwell H units. (Note that in all cases, 1 mm is etched from all surfaces after vacuum anneal to remove contamination and the equation is only for noncontaminated material.) It is believed that more detailed studies of this type could be very useful in developing engineering guidelines for the more effective use of Mo and its alloys. While much has been done to develop basic understanding of the embrittlement phenomena, little has been done to develop a sound engineering basis for using the materials. Far more work is needed to derive relations, empirical or otherwise, that effectively define the boundaries and limiting conditions for the ductile to brittle transition in the materials that will be used for nuclear space power systems.

Weld Embrittlement

Weld embrittlement can occur with all the refractory metals and alloys. In the case of Cb and Ta, embrittlement is the result of contamination with C, O, H, or N during welding. As shown in Figure 14, nitrogen has a severe embrittling effect on welds in Cb alloys. The addition of 300 ppm N_2 raises the weld DBTT from -200° to $+50^{o}F$. Oxygen is shown to have a comparatively minor effect. Similar effects are observed with tantalum, and nitrogen has by far the greatest hardening and embrittling effect as a contaminant. Hydrogen embrittlement of Ta welds also can be major problem. Unfortunately, few definitive studies that relate weld contamination to weld atmosphere and weld contamination to weld embrittlement have

been made for the major commercial alloys. Suitable engineering guidelines to control and eliminate this problem do not exist, and work should be undertaken for any proposed alloys of use to define precise relationships.

Welding of Mo and TZM-Mo presents a unique problem, but of a somewhat different nature. Weld joints tend to be brittle under most conditions. Even those made with ductile Mo-Re alloys can be brittle as a result of recrystallization in the heat-affected zone adjacent to the weld bead. Ductile welds (3-10 percent elongation at 72°F) have been made by EB welding (Kalns, 1975); however, this appears to be an exception rather than a rule.

Recent studies by Wadsworth et al. (1982b) have brought the problem into perspective and indicate possible approaches to producing consistently ductile welds. Low ductility of welds in TZM-Mo is shown to be a result of localized strain variation and not the result of material embrittlement. The difference in strength level between the weld bead, the heat-affected zone, and the base metal governs the joint ductility. Welds of fully recrystallized sheet are ductile, since strength is uniform across the joint area. Welds in stress-relieved sheet are brittle because all of the strain is localized in a weak but ductile weld bead. If the weld bead strength is increased (Mo-Re alloy weld), the deformation is then localized in the weaker recrystallized heat-affected zone, and the joint again is brittle. The approach to ductile welds requires a balance between weld composition and weld joint design to prevent localized straining in the region of the weld deposit. For example, by using a thick weld bead and thick adjoining heat-affected zone, stress within the weld could be reduced to the point where deformation on loading would be balanced or equalized between the joint and base metal. Ductile welds could be made in tube joints, for example, by upsetting the ends of tube before joining to produce a thick weld joint. Since the strength of recrystallized TZM sheet is about 60 percent that of stress-relieved TZM sheet, the joint would have to be 1.67 times as thick as the base metal for uniform straining after welding. Detailed studies of joint configuration and relative weld area strength parameters are needed to develop useful welded Mo alloy structures.

Creep Life Predictions

Design for creep requires accurate data on the time required to reach a given level of creep strain under specific conditions of load and temperature. Often the load and temperature are variable, and additive creep effects must be considered. Under ideal conditions, creep behavior is such that a long period of steady state creep occurs, as shown in Figure 18 for ASTAR 811-C. The slope of the steady state portion of the creep-deformation-time curve is the creep rate ($\dot{\epsilon}$) and can be used with various analytical expressions to describe an envelope of creep behavior in terms of stress and

temperature. A commonly used expression for Class I solid solutions
to which most refractory metals belong is as follows:

$$\dot{\epsilon} = A \ (\sigma/E)^n \exp \ (-Q/RT)$$

where $\dot{\epsilon}$ is the steady state (minimum) creep rate, E is the average
dynamic Youngs modulus, σ is the stress, n is a stress exponent
(usually n = 3 for Class I solid solutions), and T is the absolute
temperature. Such expressions can be used to calculate creep over a
wide range of conditions, although the data cannot be extrapolated
safely beyond the longest times used in deriving the rate equations.
That is, rate equations derived from 1,000 hours of data should not be
used to calculate 10,000-hour creep deformation.

Unfortunately, curves of the type shown in Figure 18 are the
exception and not the rule for refractory metals. More typical or
representative creep curves for T-111 alloy are shown in Figure 19.
There is no steady region of creep, and the creep rate ($\dot{\epsilon}$) increases
continuously with time. Titran (1974) found that some creep curves of
this type could be linearized by plotting ϵ versus $t^{2/3}$. The
steady state creep rate, $\dot{\epsilon}$, for such a material would be in hours $^{-2/3}$
instead of hours $^{-1}$.

A reanalysis of available long-term creep data on this basis has
not been done, and it is not known how universal the relation may be.
It more likely will be found that each creep curve will fit a slightly
different exponential time base, and the current approach is to use
computer plotting to linearize data. By feeding raw creep data along
with modulus and diffusivity data into a properly programmed computer,
creep relations for use as predictive engineering data can be
derived. It would be very helpful if an appropriate program could be
devised and data generated during the last 30 years be reanalyzed, to
develop a reliable engineering creep data base with broad predictive
capability. Without such a base, engineers will have only rough data
plots of the Larson-Miller type (Figures 6, 7, 9), which are of
limited utility.

Primary consideration should be given to establishing a
computerized analytical approach for creep data correlations on
refractory metal alloys. An added benefit of such an approach is the
ability to factor statistical variations in creep into the program in
order to establish reliable design guidelines. All creep data exhibit
a large amount of scatter that can be as much as 1-2 orders of
magnitude in time for a given amount of creep strain. Typical scatter
for a cobalt-base alloy (HS 188) is shown in Figure 20. A Weibull
plot of 0.2 percent creep life data for this alloy indicates that the
scatter is random in nature (B = 1) and covers over 2 orders of
magnitude (Figure 21). Creep data for refractory metals indicate a
similar behavior, and a statistically sound data base must be obtained

for reliable design. With a 10,000-hour life requirement, test costs
would be prohibitive unless a computerized test design and analysis
approach where employed.

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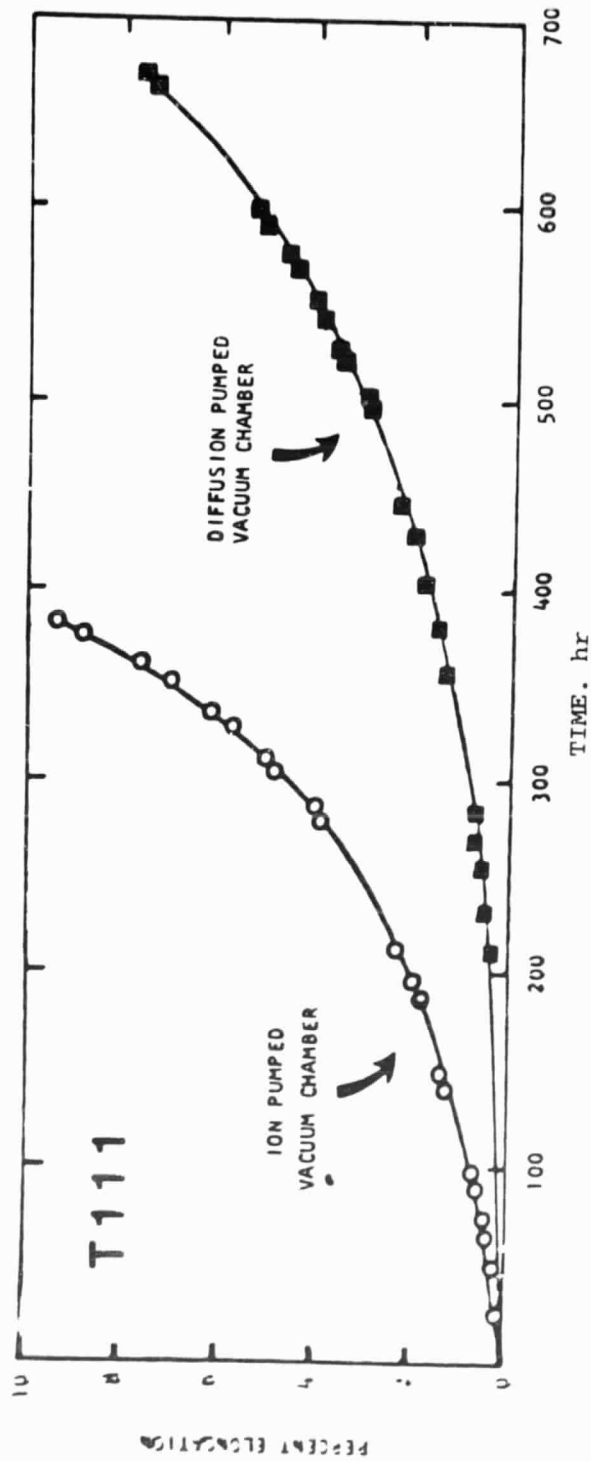
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FIGURE 1 The effect of vacuum level on creep of T-111: diffusion pumped, approximately equal to 10^{-5}, ion pumped, approximately equal to 10^{-9} (Sheffler et al., 1970).
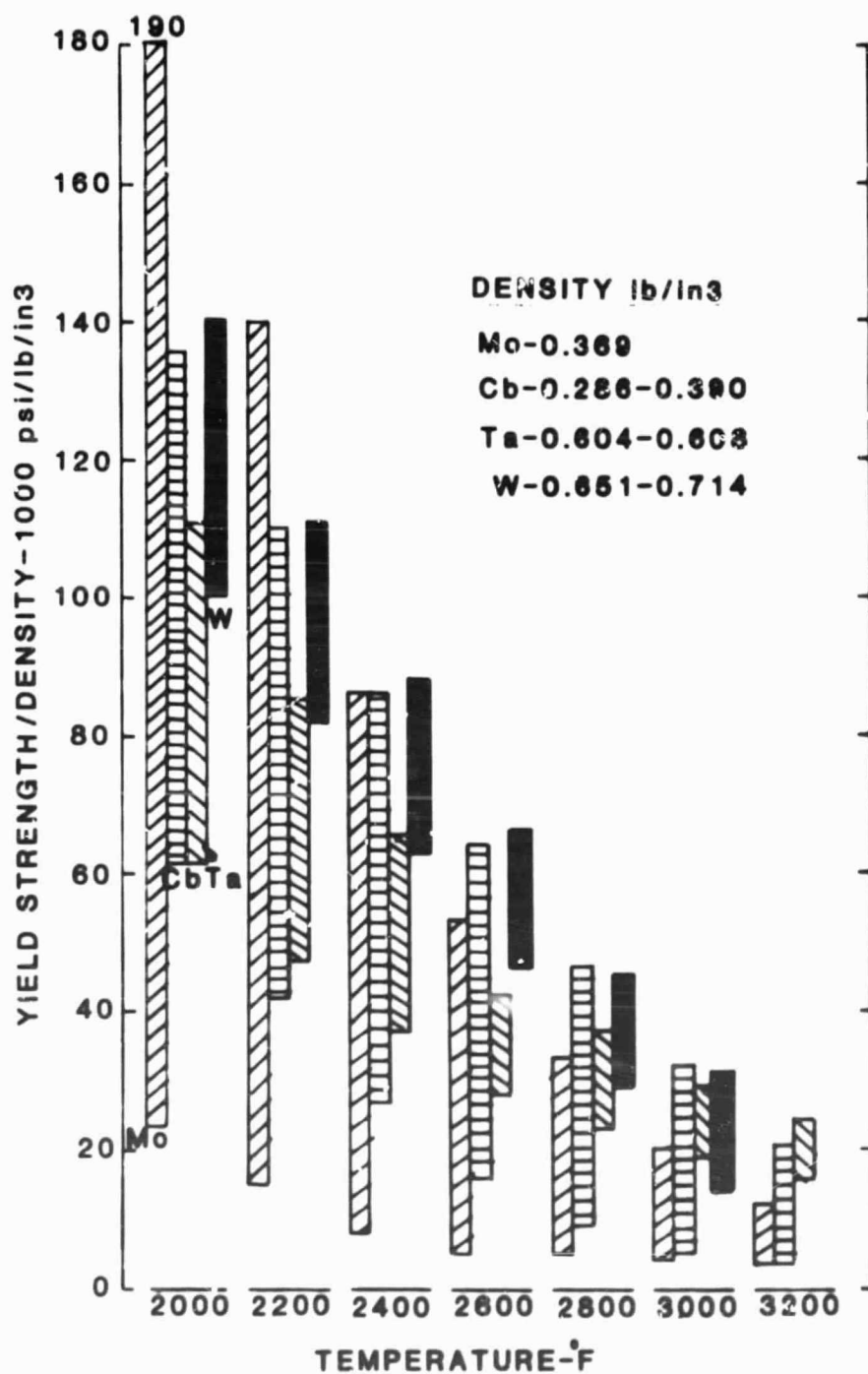
FIGURE 2 Range of yield strength (density compensated) for refractory metal alloys from 2000° to 3200°F.
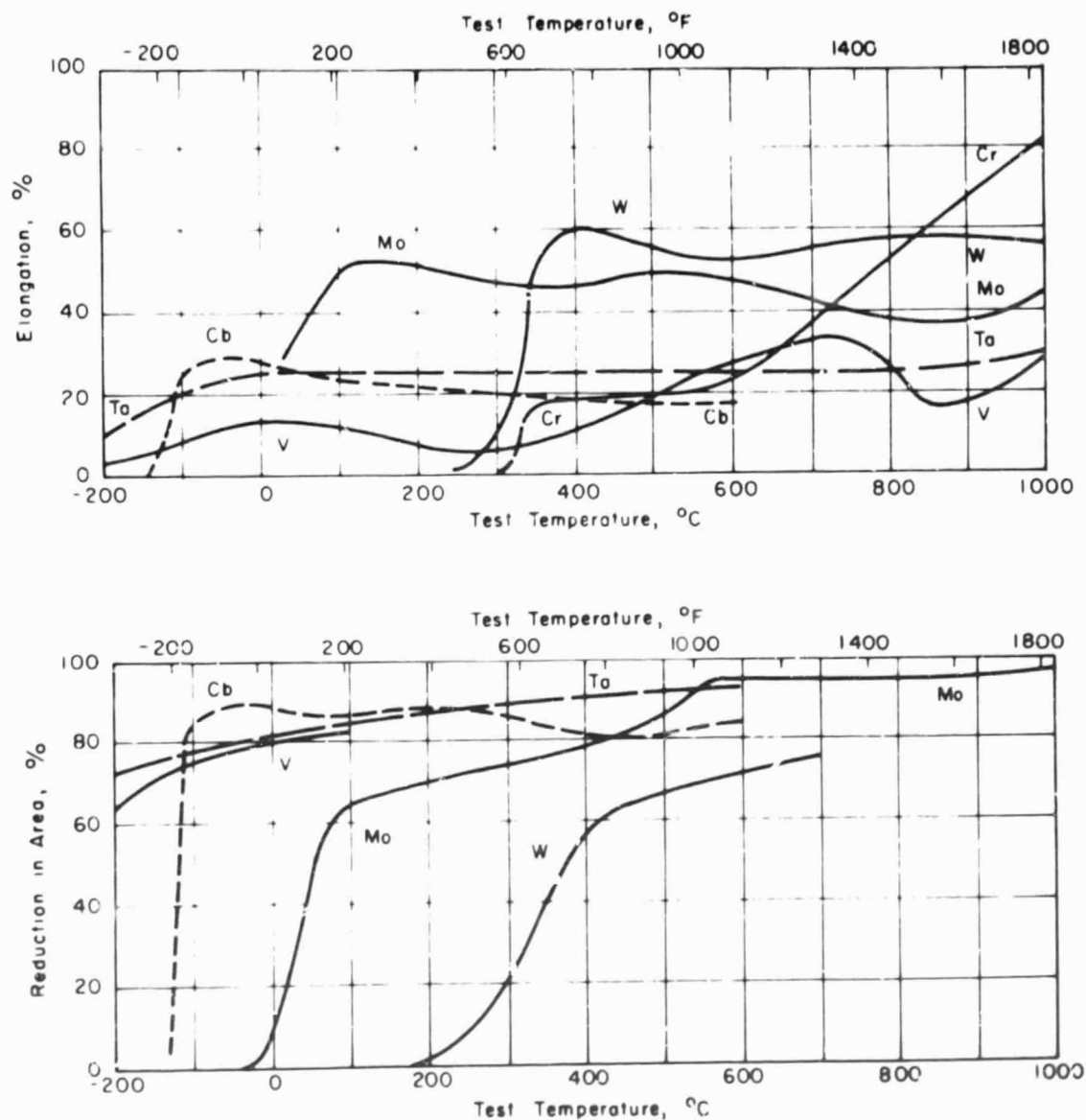
FIGURE 3 Ductile to brittle transition temperatures for refractory metals (Tietz and Wilson, 1965).

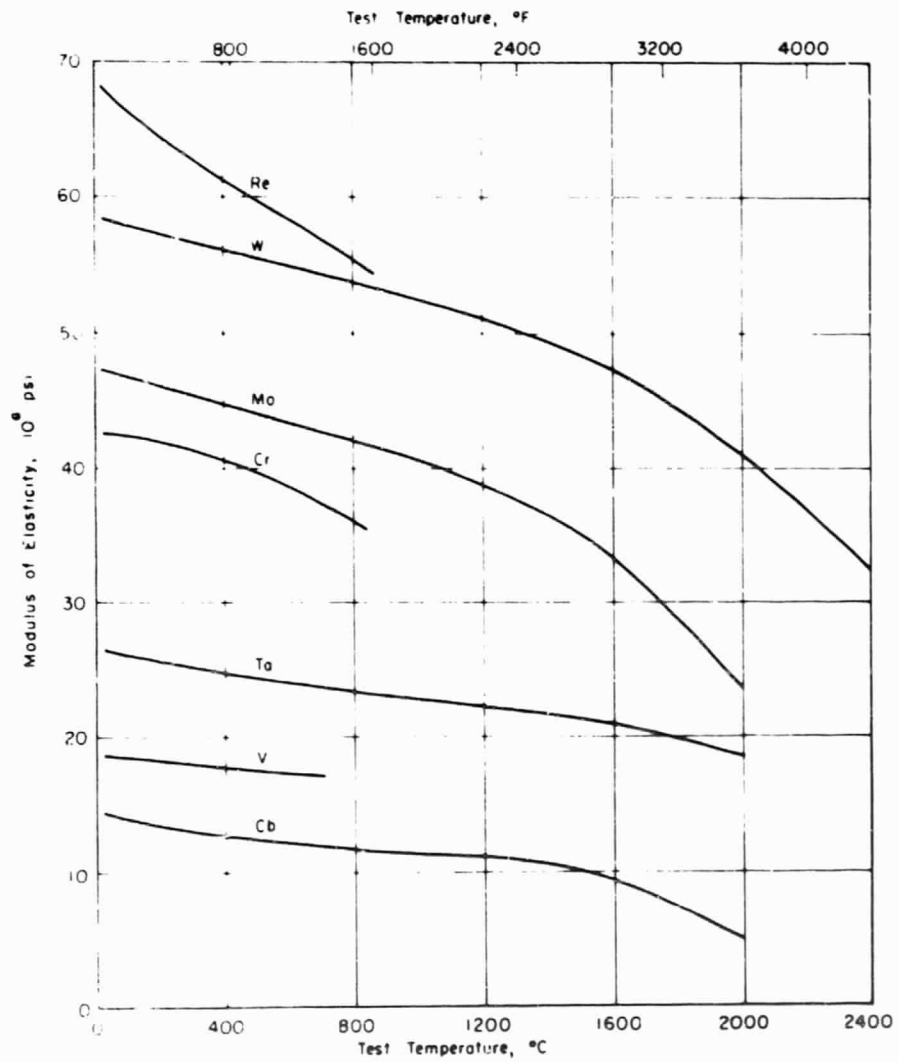FIGURE 4   Modulus versus temperature for refractory metals (Tietz and Wilson, 1965).
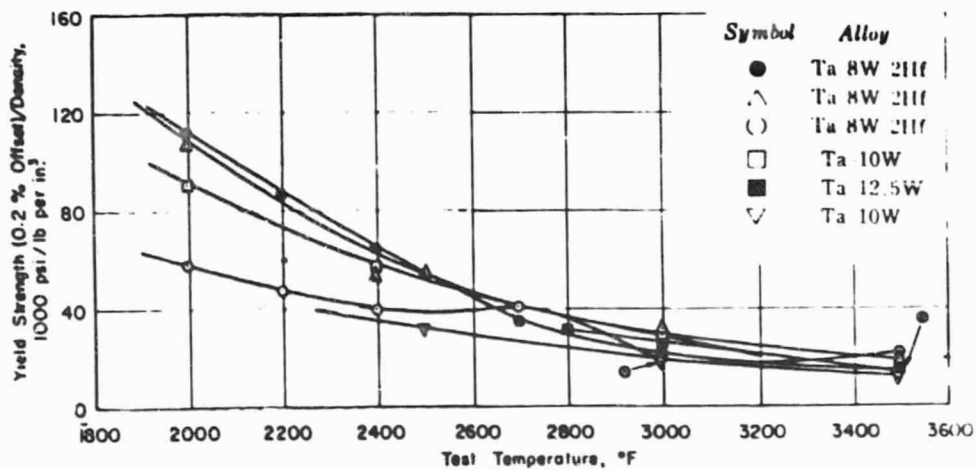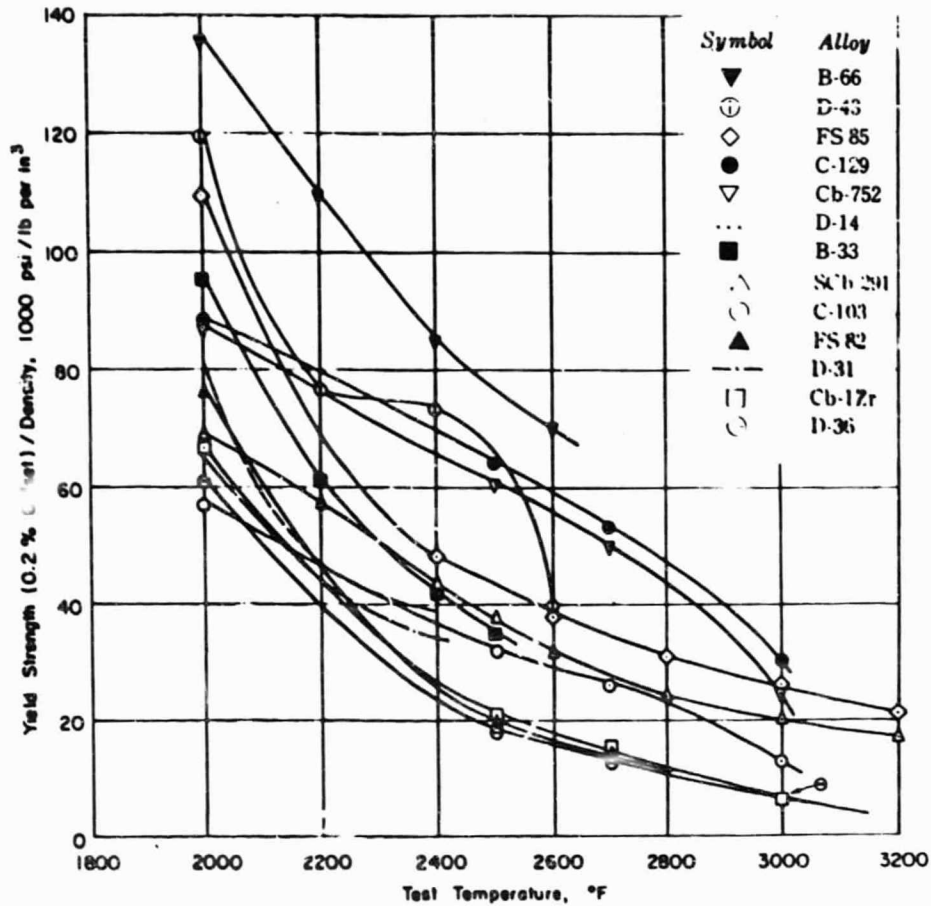
FIGURE 5 Density-compensated yield strength versus temperature for commercial Cb- and Ta-base alloys (Tietz and Wilson, 1965).
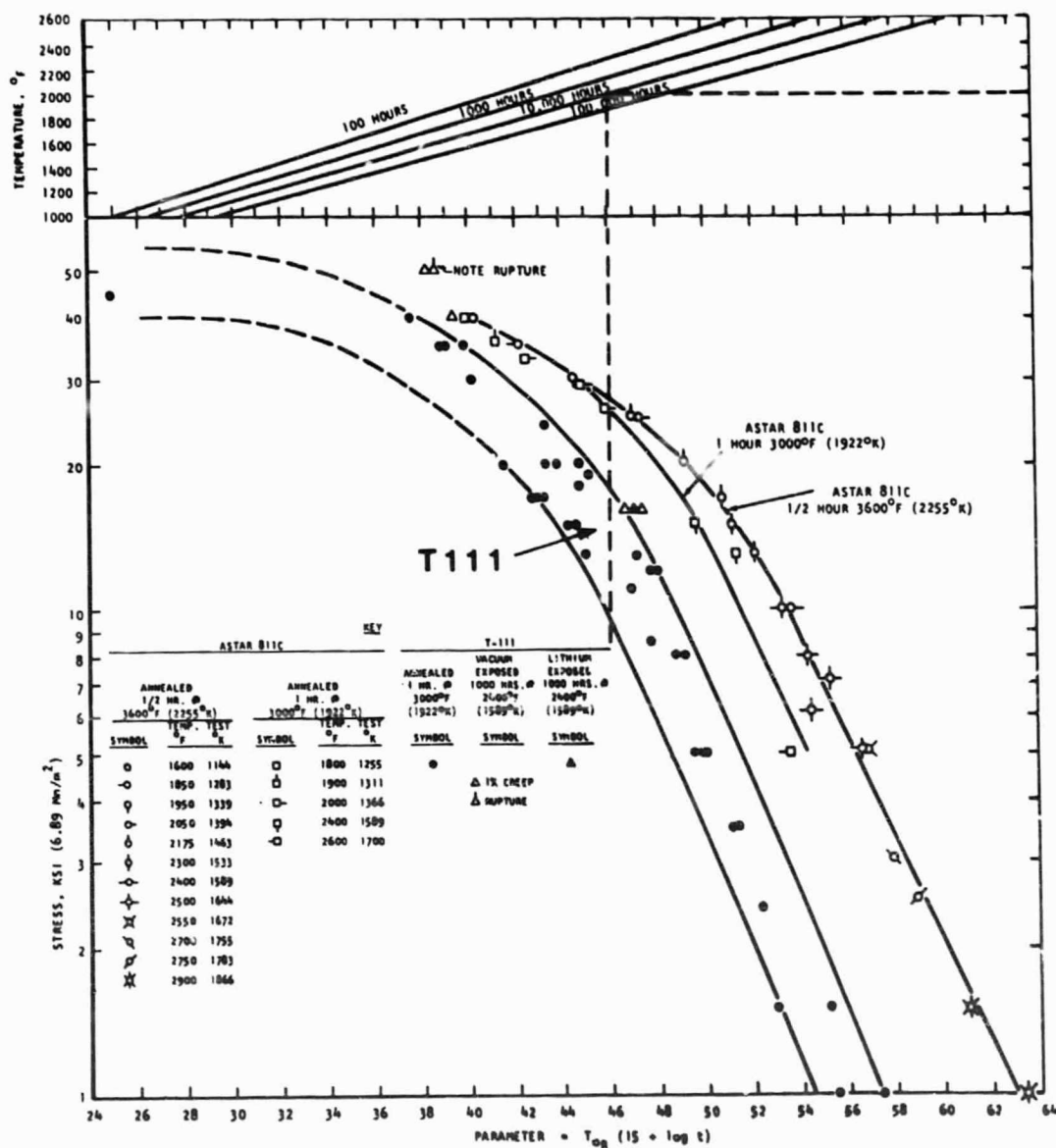
FIGURE 6 Larson-Miller plot of ultrahigh vacuum creep data for Cb-base alloys (Sheffler et al., 1970).

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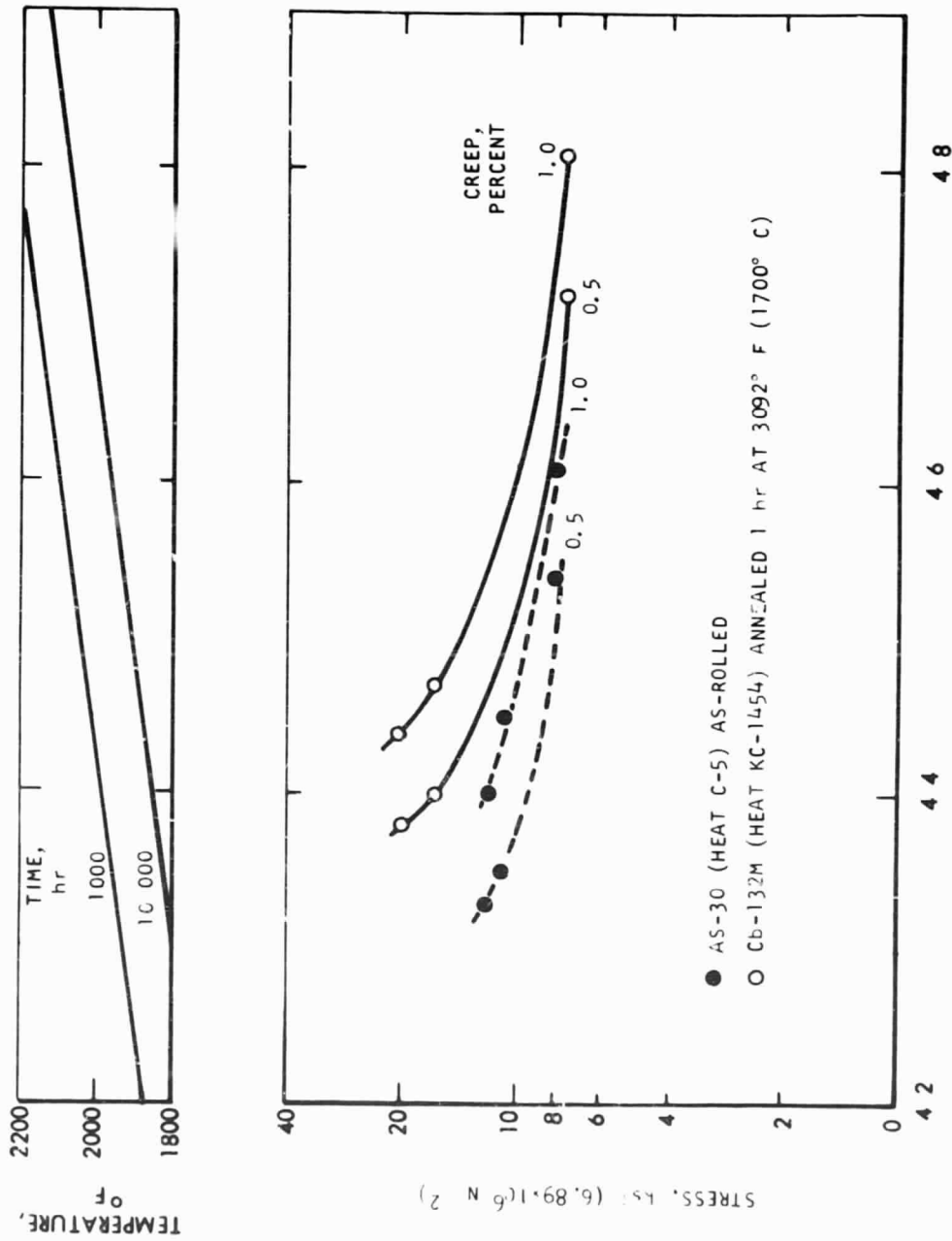
FIGURE 7 Larson-Miller plot of ultrahigh vacuum creep data for Cb-base alloys (Sheffler et al., 1970).
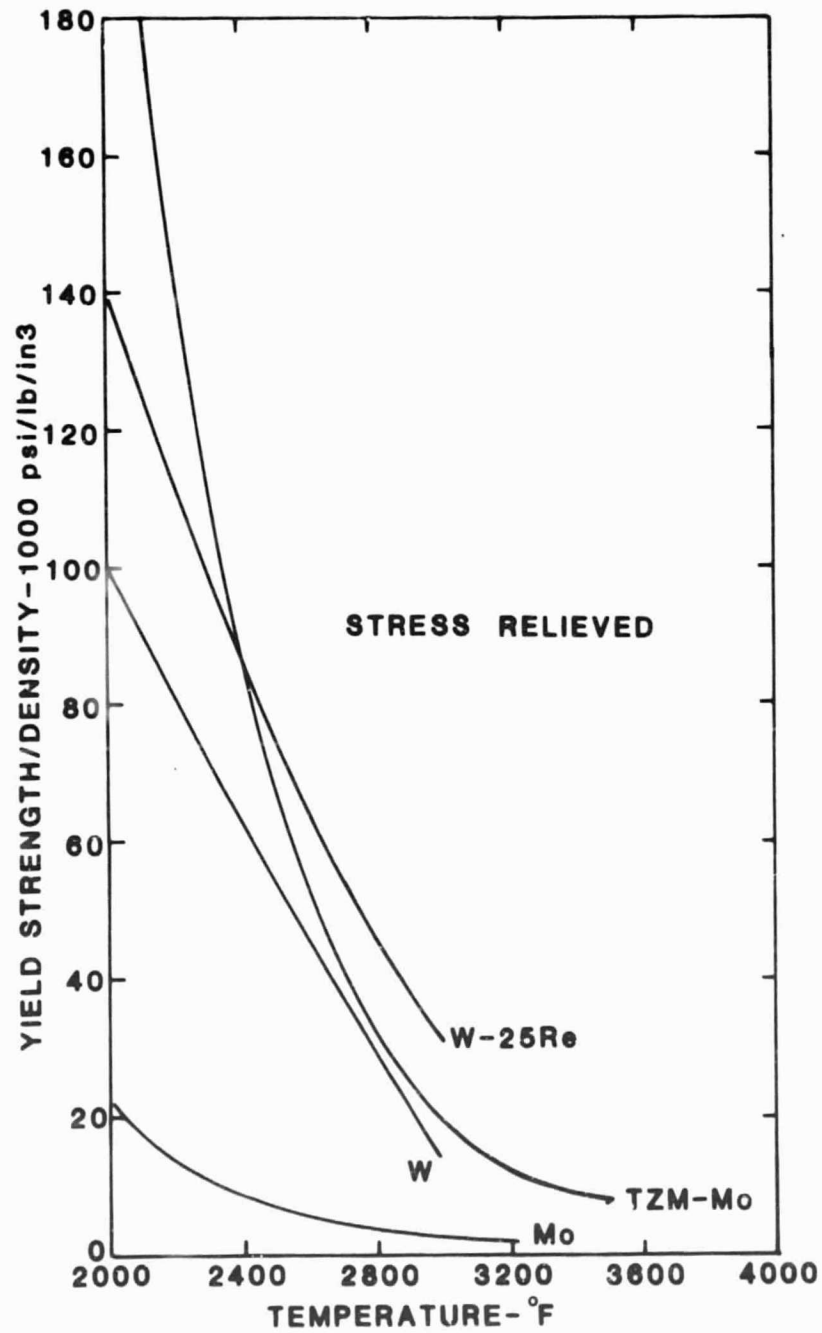
FIGURE 6 Density-compensated yield strength versus temperature for Mo- and W-base alloys (Mo: Tietz and Wilson, 1965; W: Lessman and Gold, 1970).
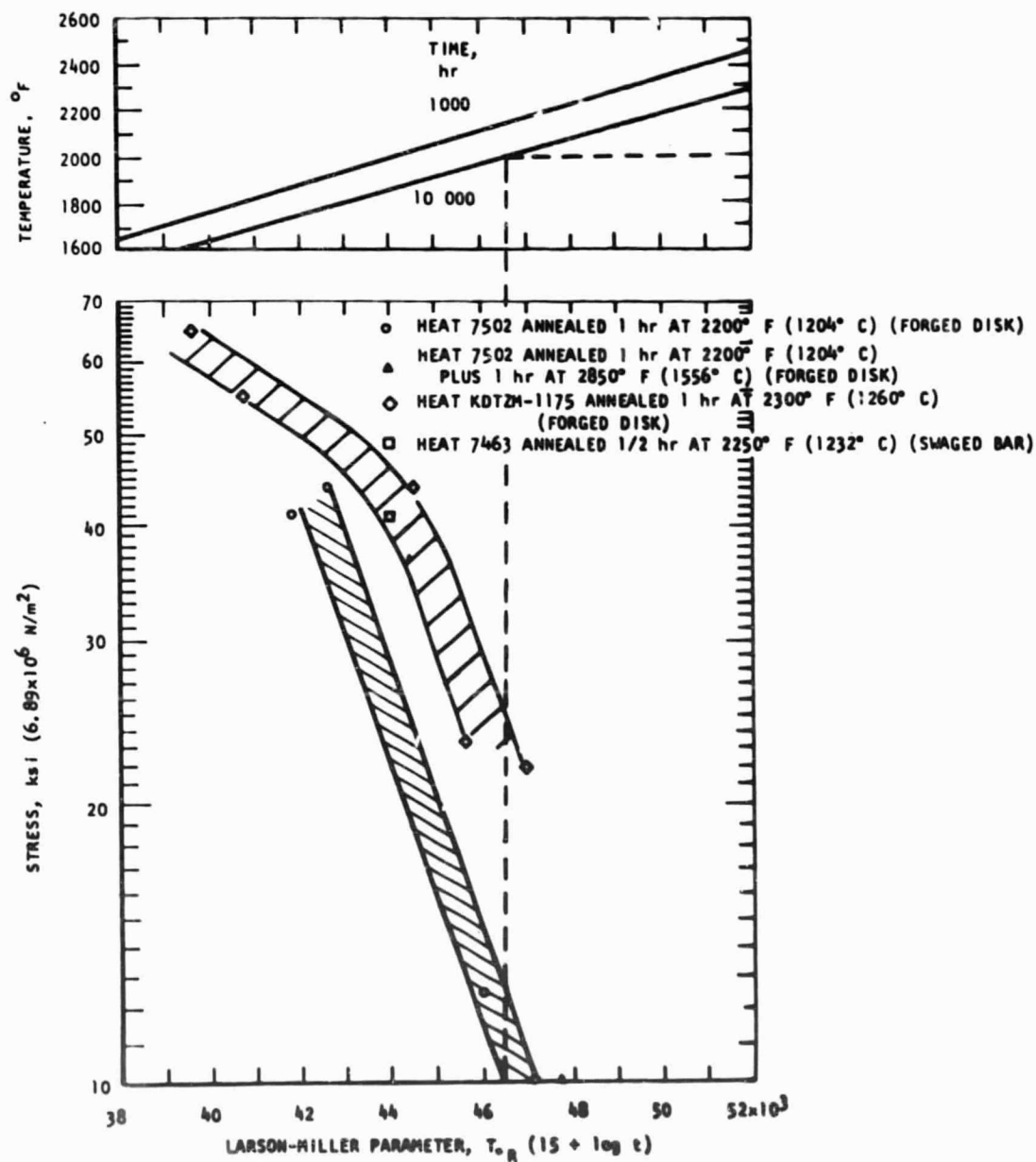
FIGURE 9 Larsen-Miller plot of ultrahigh vacuum creep data for
TZM-Mo (Sheffler et al., 1970).
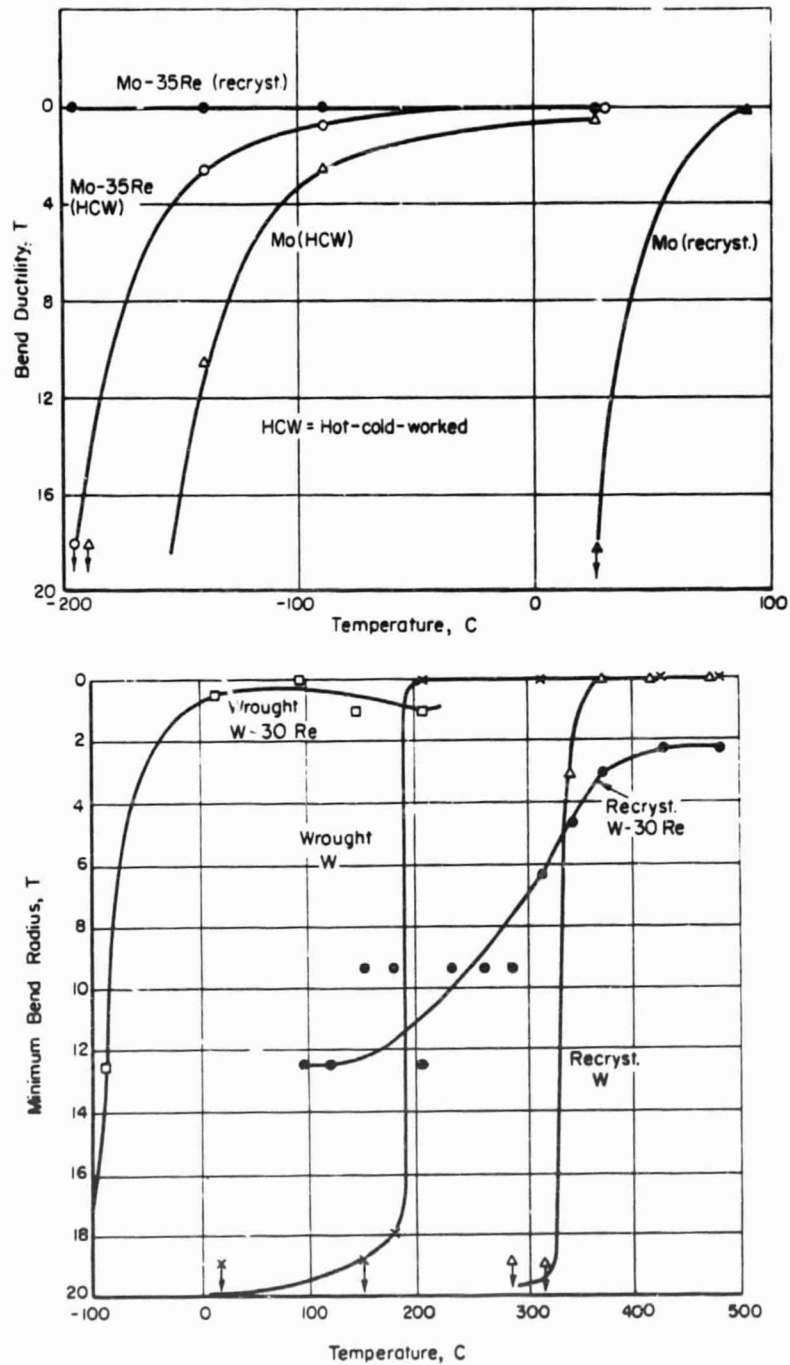
FIGURE 10   Ductile to brittle transition for Mo and W alloys (Jaffee and Sims, 1958).
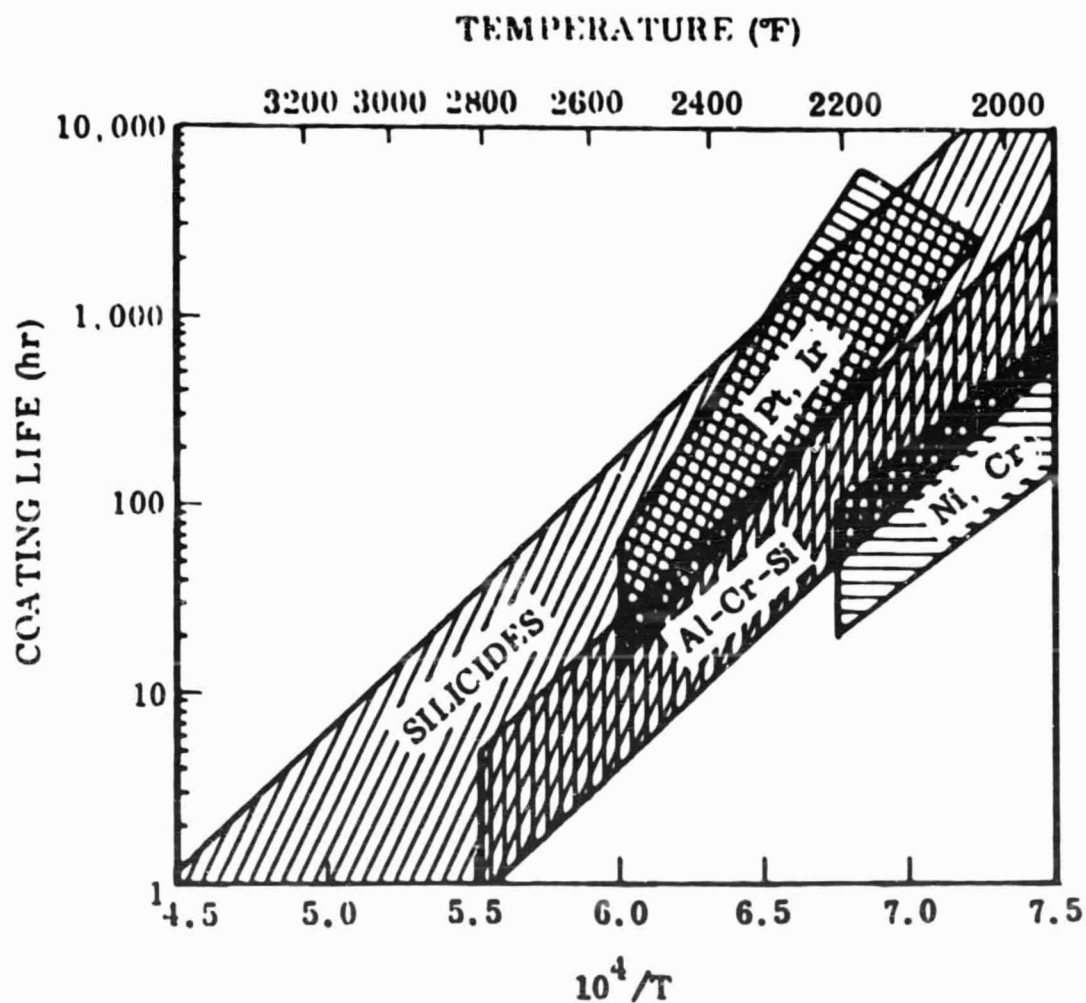
FIGURE 11 Useful life of coatings on Mo-base alloys under steady state exposure conditions in air (NMAB, 1971).

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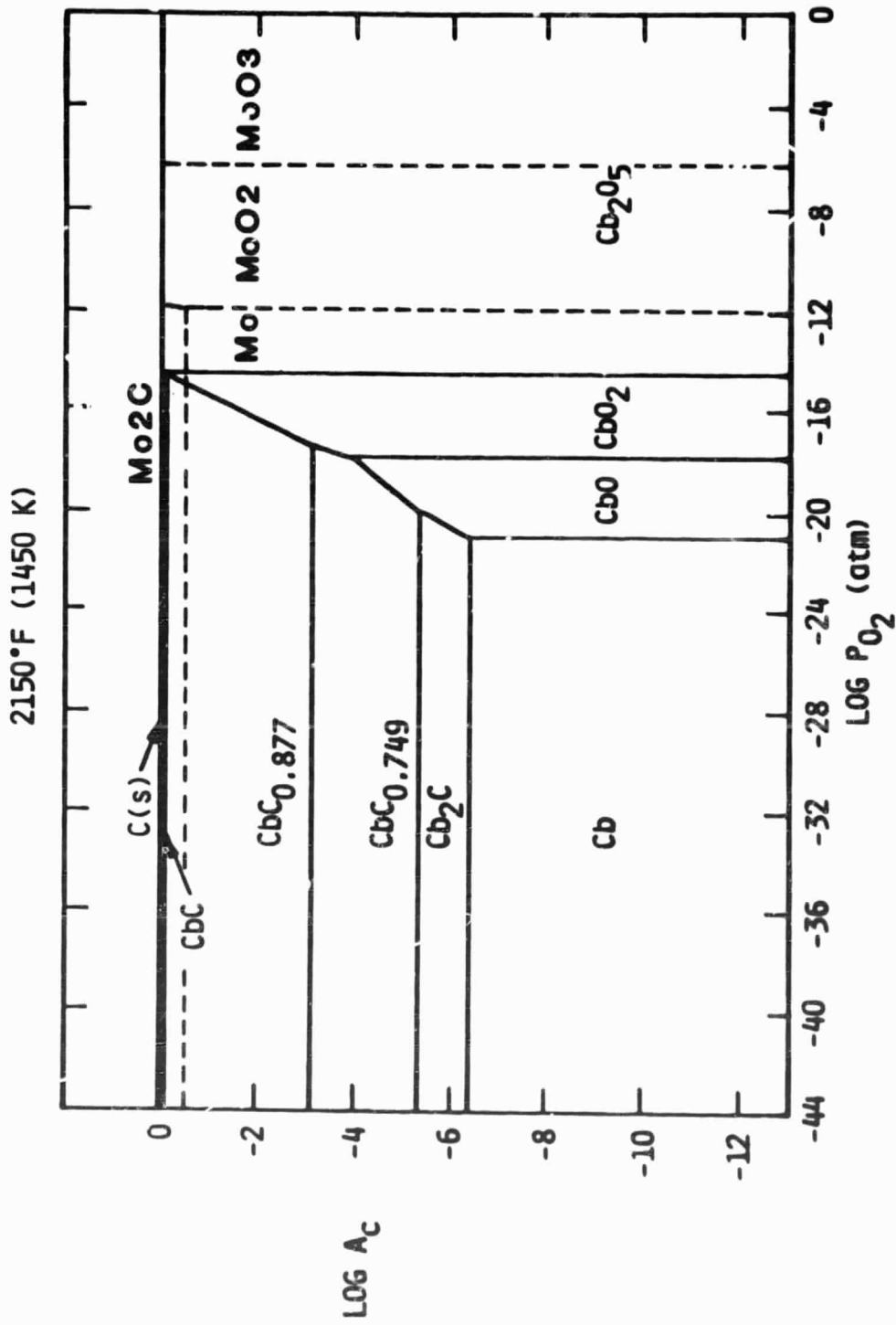
FIGURE 12   Cb-C-O and M-C-O phase stability diagrams at 2150°F (Perkins and Packer, 1981).
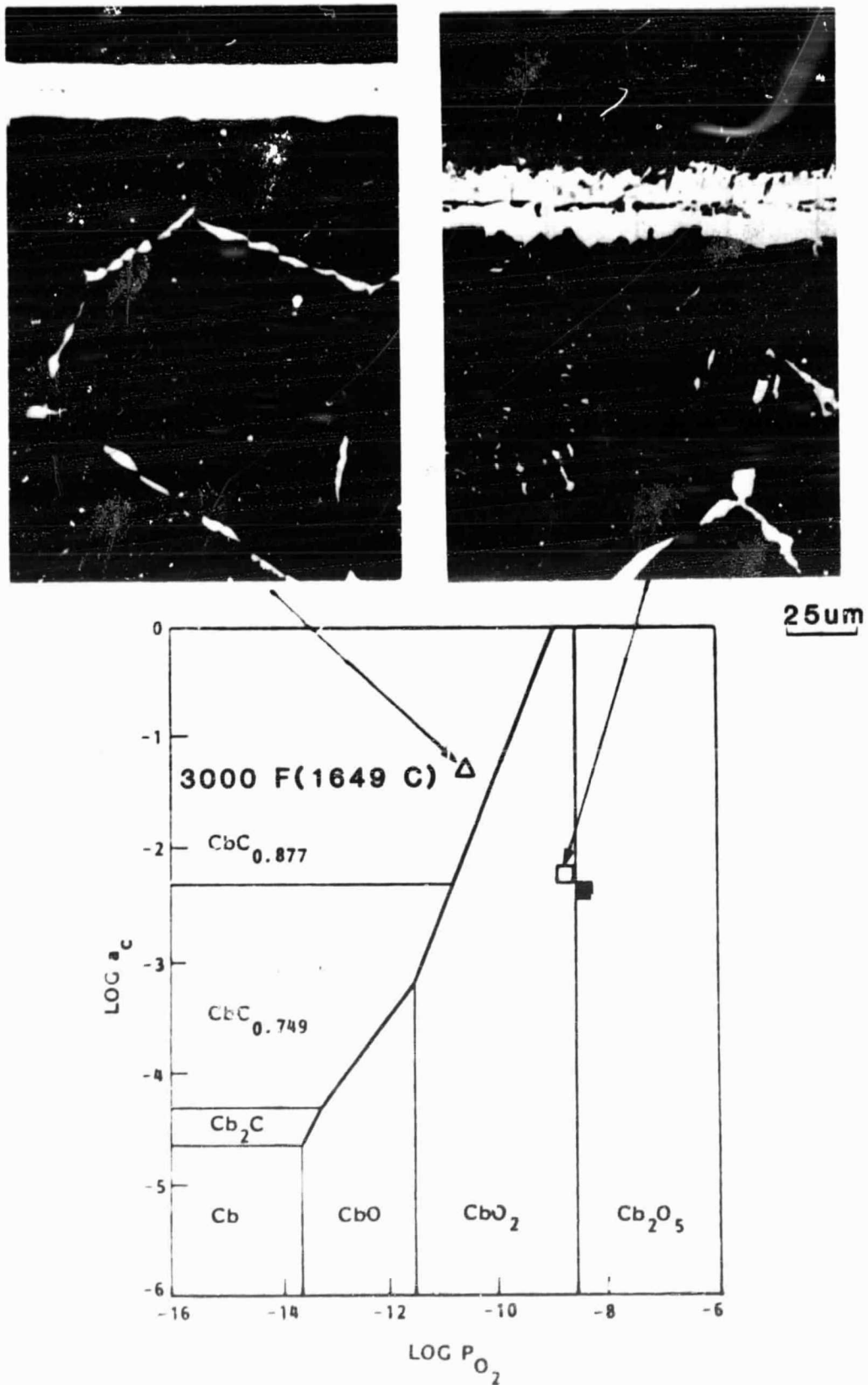
FIGURE 13 Reaction of Cb with low P_{O_2}, low a_c atmospheres, 30 min at 3000°F (Perkins and Packer, 1981).

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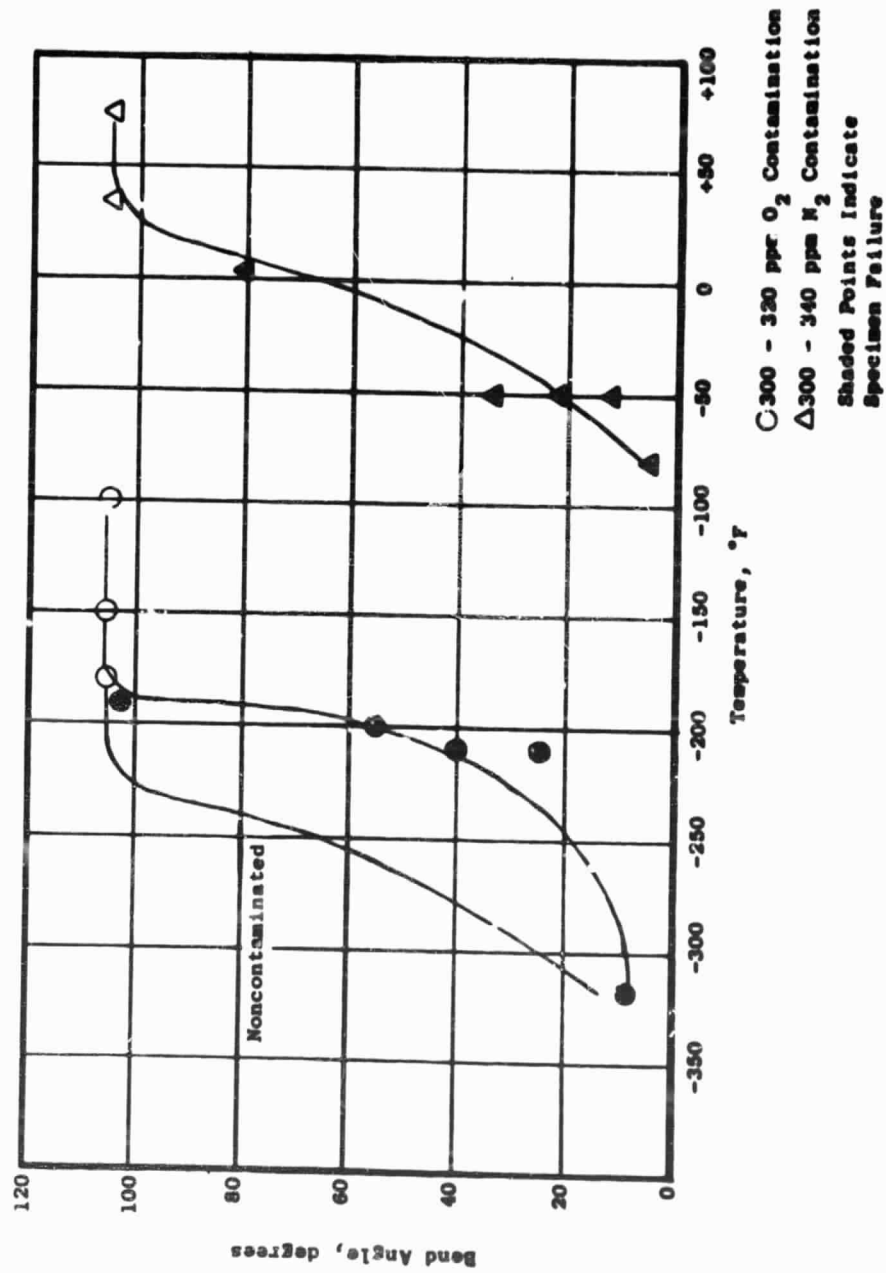
FIGURE 14 Effect of contamination on the DBTT of B-66 alloy weld metal (Thompson et al., 1966).
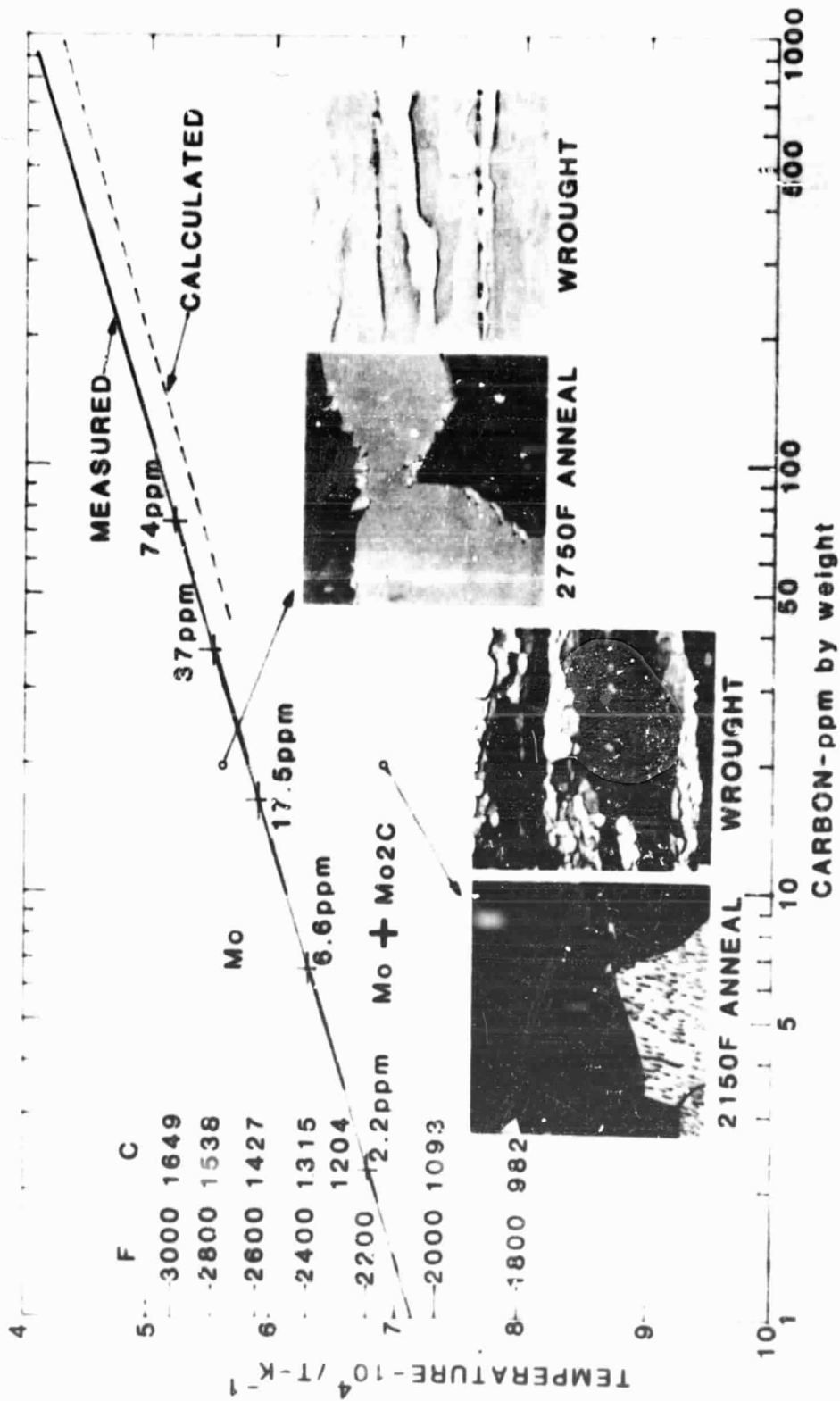
FIGURE 15 Solubility of C in Mo and precipitation of carbides during cooling.
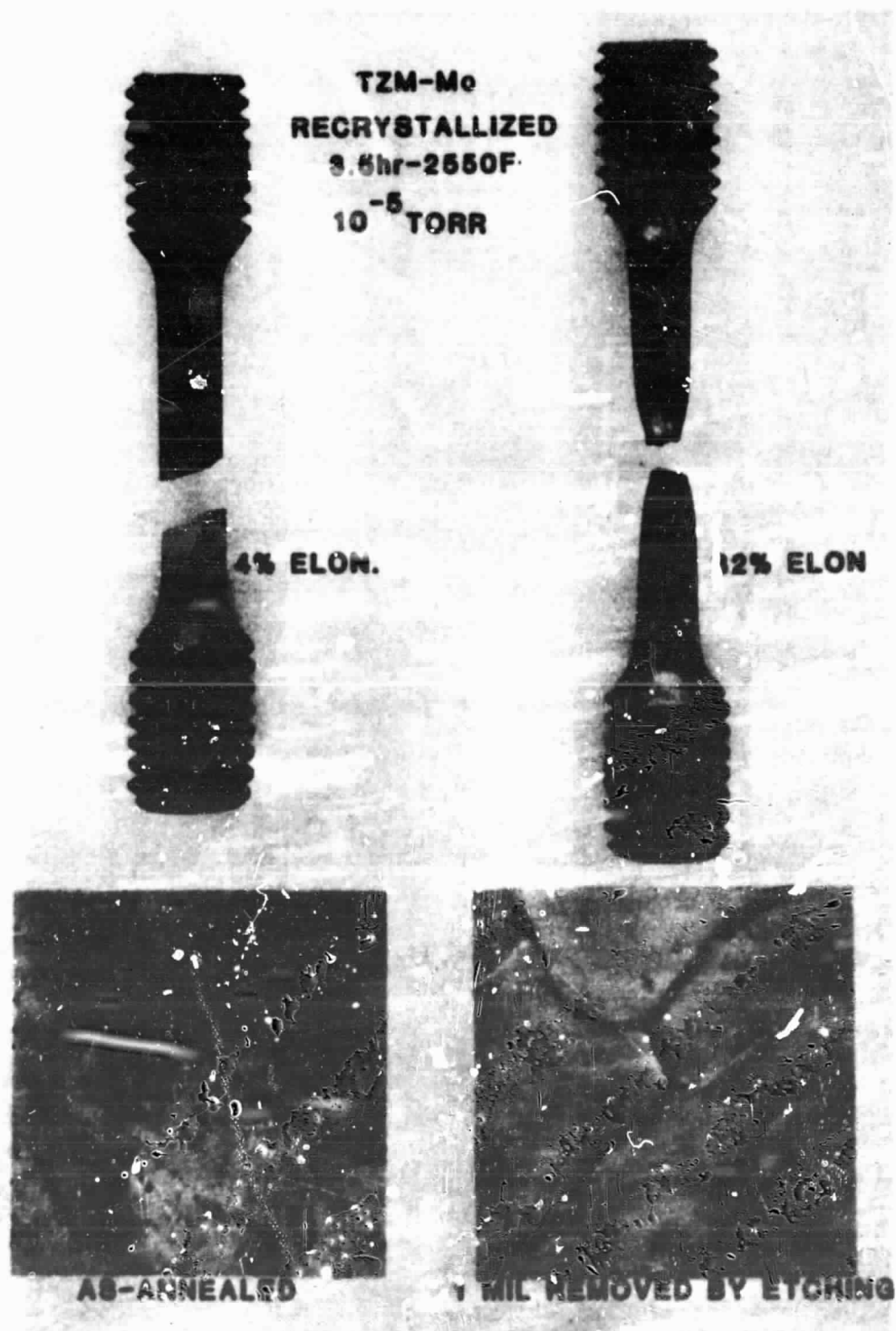
FIGURE 16 Embrittlement of recrystallized TZM-Mo by surface contamination with oxygen (Perkins and Packer, 1980).
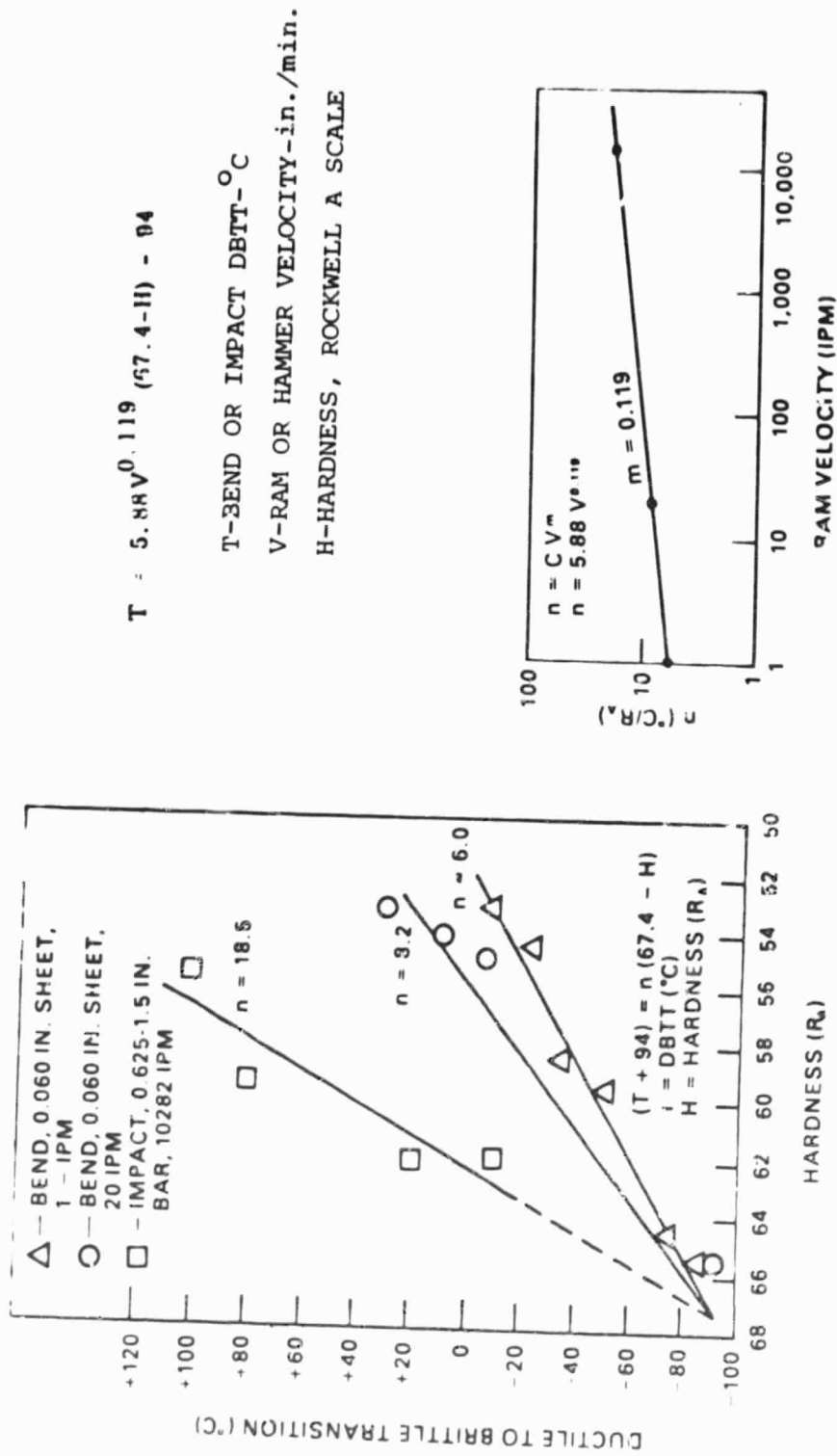
FIGURE 17 DBTT correlation with hardness and load rate for wrought and recrystallized TZM-Mo (Perkins and Packer, 1980).
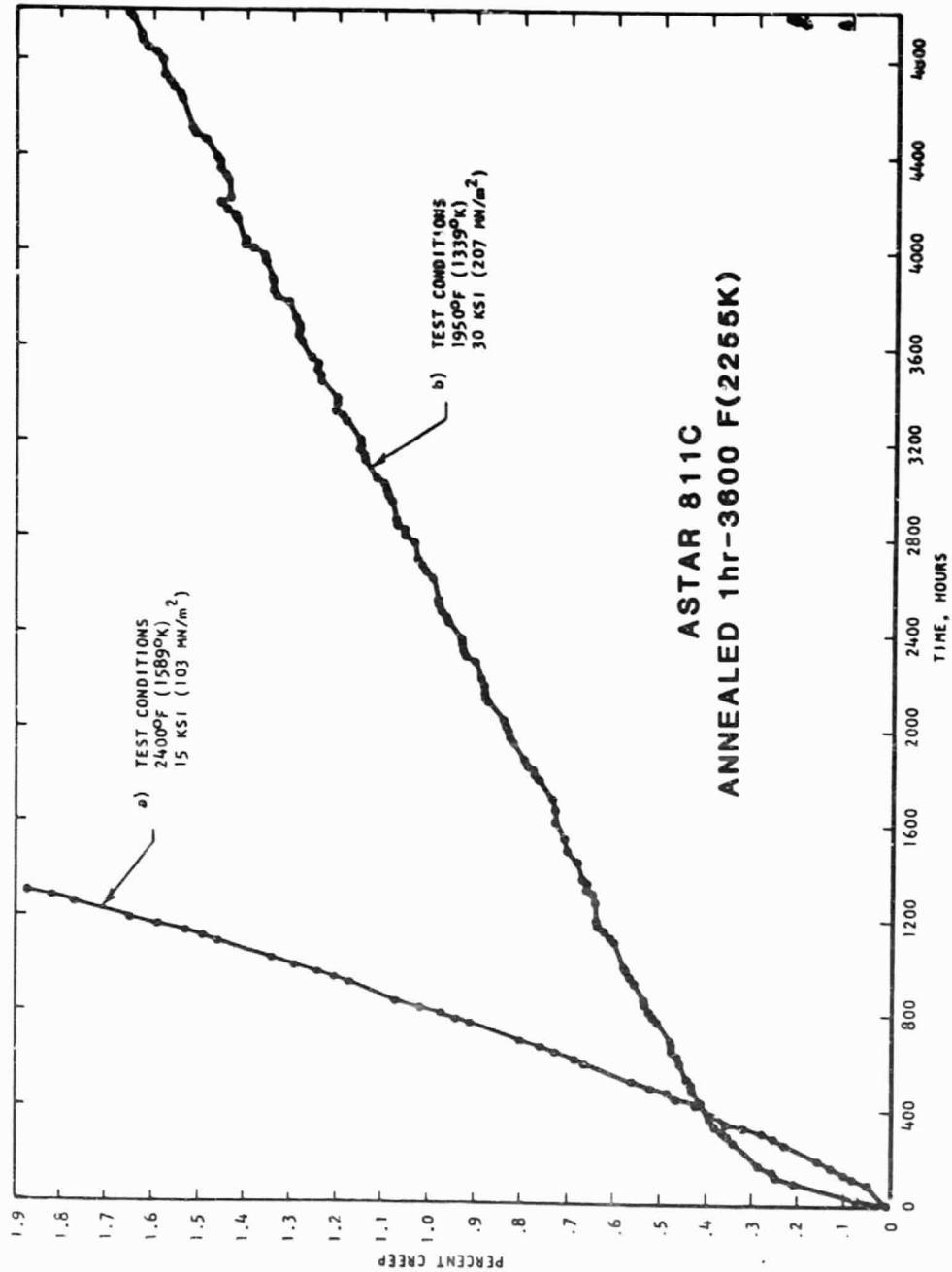
FIGURE 18 Creep curves for ASTAR 811-C in ultrahigh vacuum (Sheffler and Ebert, 1973).
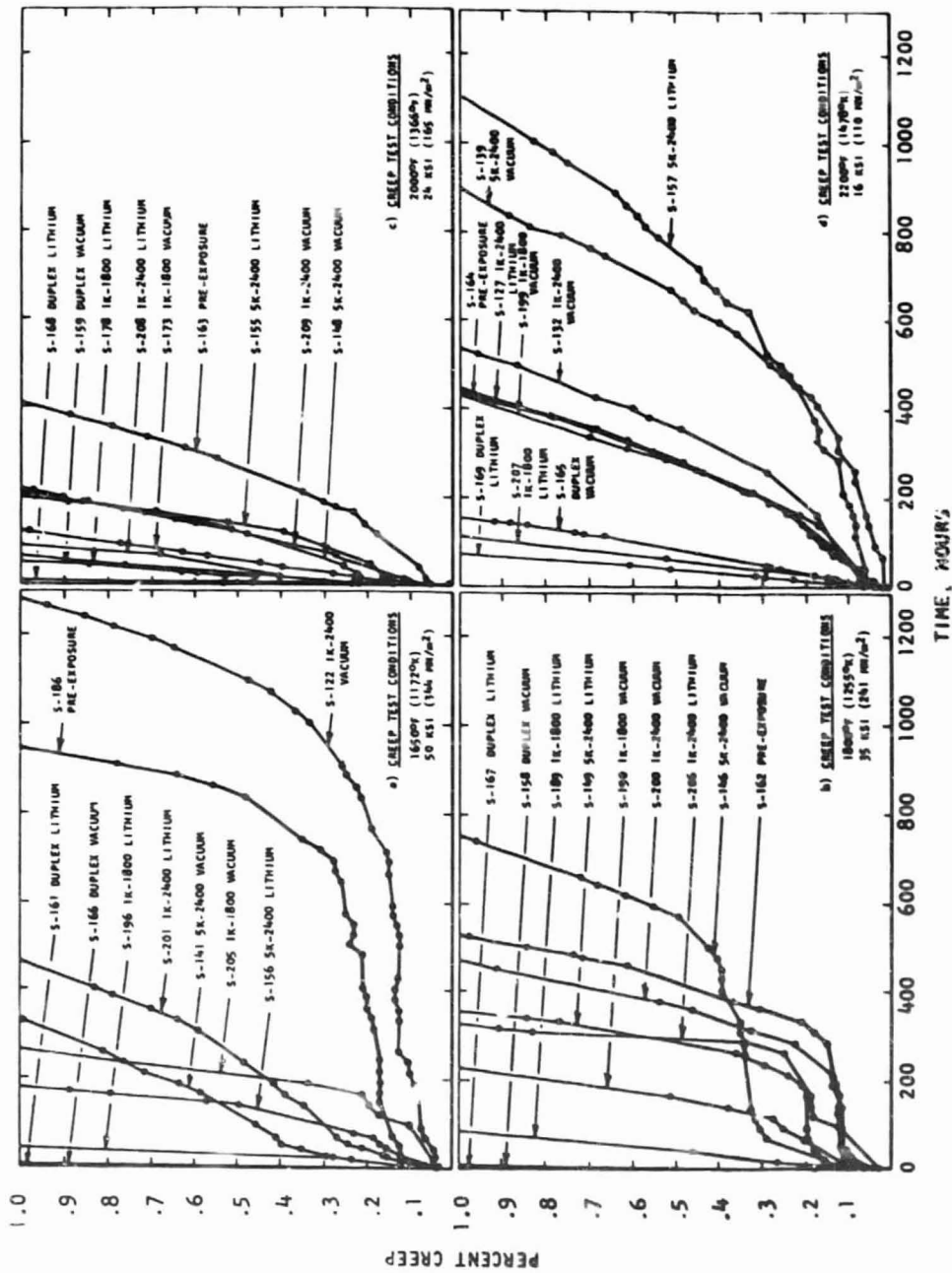
FIGURE 19 Creep curves for T-111 in lithium and ultrahigh vacuum (Sheffler and Ebert, 1973).
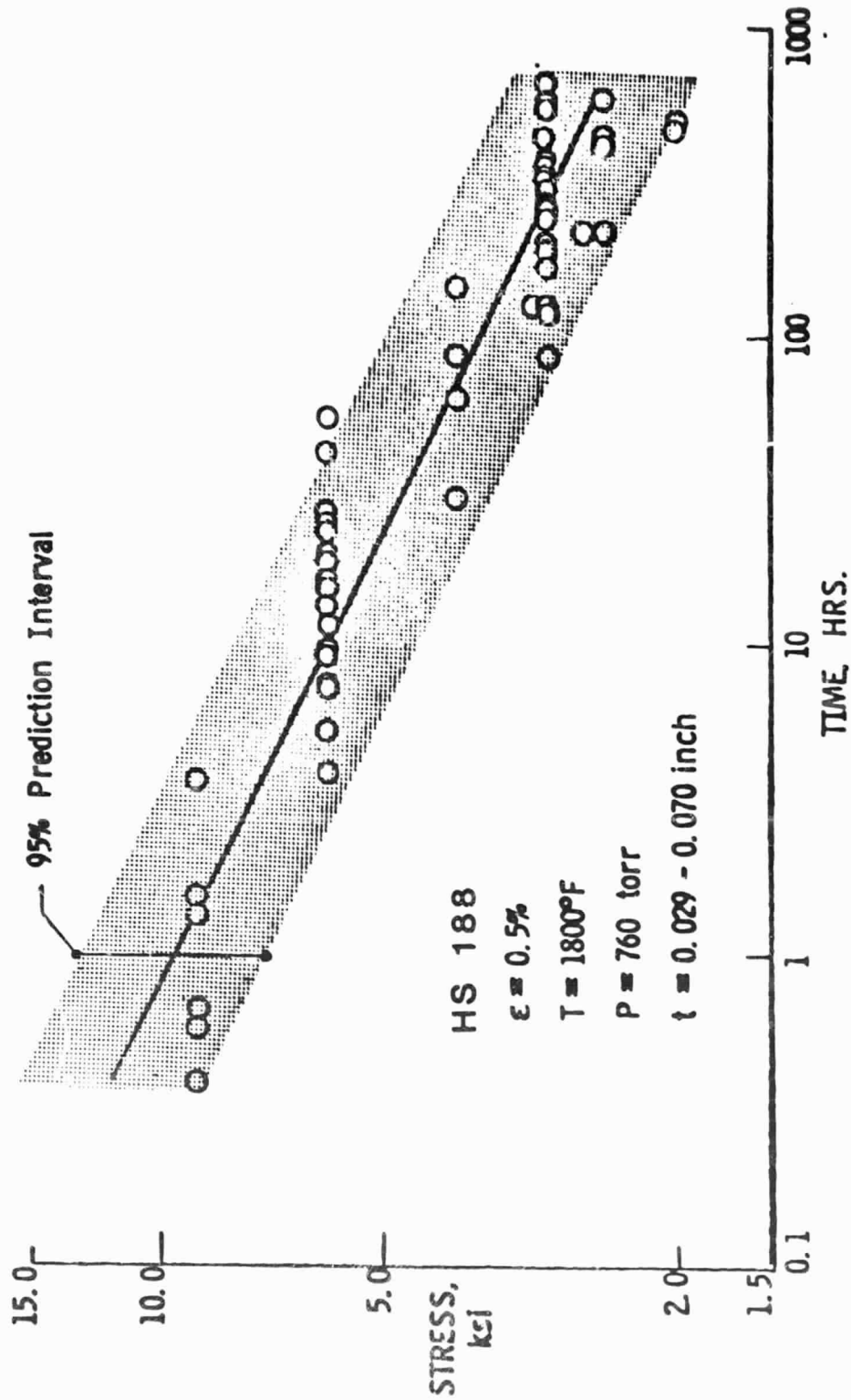
FIGURE 20 Scatter in creep strain data for HS 188 alloy (Co base) at 1800°F.
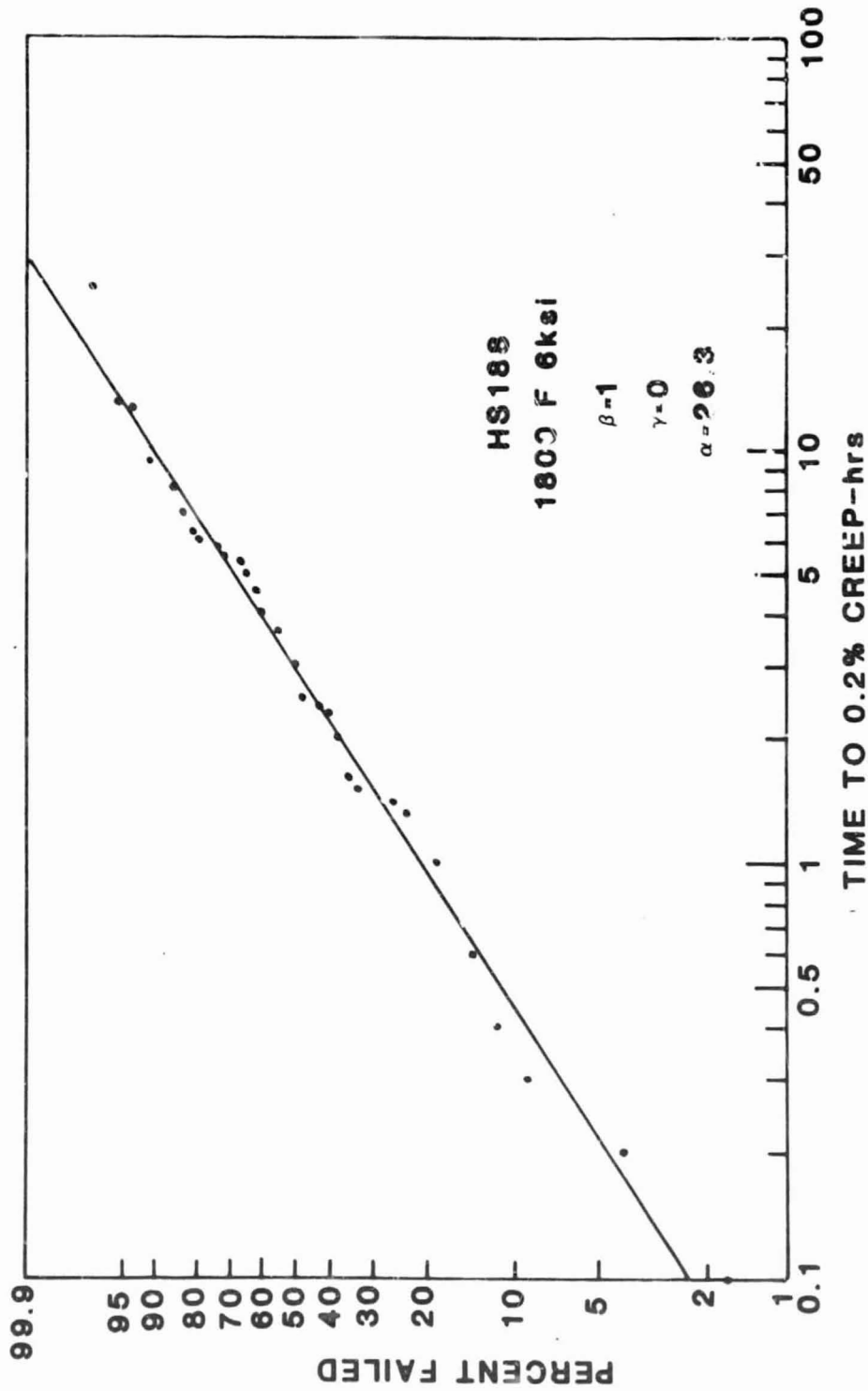
FIGURE 21 Weibull plot of creep strain data for HS 188 alloy (Co base) at 1800°F and 6 KSI.

HIGH-TEMPERATURE FUELS FOR ADVANCED NUCLEAR SYSTEMS

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ABSTRACT

High-temperature nuclear fuels are reviewed in detail, including prior
history of use, fundamentals of performance, and characteristics of
specific fuel types. It is shown that while UO_2 is by far the most
widely used fuel, UC and UN offer significant improvements in thermal
conductivity and in fuel density. However, these improvements are
attainable only at the cost of significant additional development.
Coated-particle fuels potentially offer the best high-temperature
performance and offer wide flexibility in the selection of materials
for compatibility with various coolants and core structural
materials. Many factors such as thermal properties, radiation
performance, chemical compatibility, fuel density, and development
status must be considered in the selection of fuel for an advanced
nuclear system. The selection for a particular application will
involve a tradeoff of systems requirements and fuel characteristics.

INTRODUCTION AND BACKGROUND

In nuclear systems a wide variety of fuels has been used, including
the metallic form, hydrides, oxides, carbides, nitrides, and cermet
forms of uranium and plutonium. The choice of fuel for a particular
system depends on a variety of factors, of which temperature
capability is only one. Others include chemical compatibility with
cladding coolant, and other core materials; thermal stability; and
radiation effects such as swelling and creep. While the term
"high-temperature fuel" is not well defined, it is generally taken to
include all of the uranium fuels except the metal and the hydride. In
general, the plutonium-based fuels have lower melting temperatures and
are less suitable for high-temperature applications than uranium fuels.

The term "high-temperature fuel" is somewhat misleading, since in
all systems involving solid fuel the important temperature is that of
the coolant exiting the core. There are may cases in which the peak
fuel temperature in one system may be higher than that in another

system, but the outlet coolant temperature is lower. A good example of this is the commercial Light Water Reactor (LWR) versus the High-Temperature, Gas-Cooled Reactor (HTGR). The high outlet temperature and consequent high efficiency of the HTGR does not come from a high peak fuel temperature but comes from the high surface temperatures allowed by the fuel particle coatings and inert coolant. Thus the fuel cannot be considered in isolation from the remainder of the system. Some of the fuels that have been proposed for space nuclear applications are shown in Table 1.

FUNDAMENTALS OF NUCLEAR FUEL BEHAVIOR

A general understanding of the fundamentals of nuclear fuel behavior is essential to a discussion of fuel selection for advanced nuclear systems. A number of aspects of the performance of nuclear fuel that are generic, in that they apply to all fuel types under the appropriate conditions, are discussed in this section. This includes primarily irradiation and thermal effects such as fuel restructuring, swelling, and fission gas release. Much of the discussion in this section is derived from the excellent text by Olander (1976).

Restructuring in a highly rated mixed-oxide fuel pin is illustrated in Figure 1a (O'Boyle et al., 1969). It has resulted in the formation of three distinct zones in the fuel plus a central void. The central void is created by the migration of the porosity in the initial fuel material to the center of the fuel pin under the influence of the temperature gradient. The inner fuel region is characterized by dense columnar grains formed by the thermal-gradient-driven migration of lenticular pores. The mechanism for lenticular pore migration is vaporization and condensation of the UO_2. Consequently, columnar grain growth can only occur at relatively high temperatures (higher than $1700^{\circ}C$), at which the vapor pressure of UO_2 is appreciable. Outside of the columnar grain region is found a ring of large equiaxed grains. In this region, conventional diffusion-controlled grain growth occurs. This equiaxed grain region is important in that diffusion-controlled processes such as fission gas swelling and creep can readily occur. In the outer ring, fuel temperatures are too low for restructuring to occur, and there is no change from the as-fabricated microstructure. The various fuel zones and their temperatures of formation are illustrated schematically in Figure 1b (Olander, 1976).

Approximately 25 percent of the products of uranium fission are the inert gases krypton and xenon. These gases are essentially completely insoluble in the fuel matrix, so they are either released and contribute to the atmosphere surrounding the fuel, or they precipitate as bubbles within the fuel. Fission gas bubbles in nuclear fuel behave in an analogous manner to the more familiar soap bubbles. The pressure of gas in the bubble (P) is balanced by the surface tension (δ) plus the hydrostatic component of externally applied stress (σ),

TABLE 1 Materials Choices for Selected Space Nuclear Programs

System	Fuel	Cladding	Coolant
SNAP series	U-ZrH	Stainless steel	NaK, Li
Rover/NERVA	UC_2	Pyrolytic carbon	H_2
SPUR/SNAP-50	UN	Nb-1% Zr	Li
In-core thermionic	$UC-ZrC, UO_2$	W	Na
Brayton (Garrett-AGN)	UO_2	Hastelloy X	He
Brayton (GE)	UO_2-W	Refractory metals	He
SP-100	UO_2	Mo-Re	Heat pipes
Rotating bed	$UC_2, UC-ZrC$	Pyrolytic carbon	He
Fixed bed	UC_2	Pyrolytic carbon	He

SOURCE: Layton et al. (1982).

so that an equilibrium size (r) is established according to the classical expression $P = (2 \delta/r) + \sigma$. This is illustrated in Figure 2. An addition of fission gas to the bubble or a change in the applied stress will cause the bubble to grow or shrink by the addition or loss of vacancies in order to maintain equilibrium.

Since the gas in the bubbles occupies a much larger volume than the equivalent amount of gas dispersed as individual atoms in the matrix, the precipitation of fission gas into bubbles results in a volume increase, or swelling of the fuel. Bubbles can migrate, grow, and coalesce both by diffusional processes and by interaction with fission tracks. Large bubbles can also grow by plastic deformation processes. It is clear from the bubble equation that as a bubble grows, its internal pressure diminishes. Consequently, bubble growth results in an increase in swelling. Fortunately for many applications, bubble growth and swelling do not continue indefinitely. Large bubbles tend to form at the grain boundaries, where they can link up and release their gas to the surrounding environment. Thus the phenomena of swelling and fission gas release are inextricably interrelated.

A comparison of swelling rates of oxide, carbide, and nitride fuels is given in Figure 3 (Bauer, 1972). All of the fuel types follow the same trends, as would be expected. These swelling data should be used with caution, since they are valid only over a limited range of volume change and in some cases were obtained from measurements on cladding rather than fuel.

In a practical sense, swelling, fission gas release, and columnar grain growth can result in substantial shape changes and redistribution of the fuel, as illustrated schematically in Figure 4 (Olander, 1976). In stage 2 of this illustration, the fuel swells

uniformly owing to the effect of fission gas bubbles. Under conditions in which columnar grain growth occurs, as in stage 3, the fission gases will be swept up and released, leaving redensified but displaced fuel, as shown in stage 4. The process can then repeat itself. This type of process can lead to substantial creep deformations of the surrounding clad or core structure that are difficult to predict or treat in design. While these processes are generally related to conventional metal-clad fuel pins, they can also occur in less conventional structures such as the SP-100 plate fuel if the temperatures and temperature gradients are sufficiently high. Fortunately, in the current SP-100 design, the conditions do not appear to be sufficiently severe to result in significant fuel redistribution by this mechanism. However, the possibility should be considered in advanced designs and under faulted conditions.

HIGH-TEMPERATURE NUCLEAR FUELS

This review will be limited to solid fuels and thus will exclude liquid, gaseous, and plasma fuels even though these have been propsed for some high-temperature systems. The categories of fuels to be considered include oxides and cermets, carbides, and nitrides. Since coated particles represent a radically different fuel concept, they will be considered in a separate section.

Oxide and Cermet Fuels

UO_2 is by far the most widely used, the most studied, and the best understood of the high-temperature nuclear fuels. It is the standard fuel for the LWR, Advanced Gas-Cooled Reactor (AGR), and Fast Breeder Reactor (FBR), as well as the Space Power Reactor (SP-100) under development at Los Alamos National Laboratory (LANL). It is generally isolated from the reactor coolant by a stainless steel or zircalloy cladding. However, in the case of SP-100, it is used as unclad wafers separated by thin Mo-Re spacers.

The high melting temperature, compatibility with metallic cladding materials, irradiation stability, and ease of fabrication of UO_2 have made it the front-runner among the high-temperature fuels. The primary disadvantages of UO_2 relative to other high-temperature fuels are its low thermal conductivity and relatively low uranium density. (See Table 2). The thermal conductivity of UO_2 is lower by a factor of 7-10, and the uranium density is lower by as much as 30 percent than those of the other high-temperature fuels.

Control of stoichiometry is a key factor in the use of UO_2 fuel. This is illustrated in the uranium-oxygen phase diagram in Figure 5 (Latta and Fryxell, 1970). Substoichiometric UO_2 can contain low-melting uranium metal that is unacceptable both from the standpoint of chemical compatibility and from the standpoint of

TABLE 2 Selected Properties of High-Temperature Nuclear Fuels

Fuel Type	Melting Temperature ($^{\circ}$C)	Thermal Conductivity W/cm-K (to 1000°C)	Uranium Density (g/cm^3)
UO_2	2865	0.03	9.5
UC	2325	0.2	13
UC_2	2480	0.35	11.7
UN	~2800 (at 2.5 atm)	0.2	13.5

irradiation performance. The chemical potential of oxygen increases sharply as the UO_2 becomes hyperstoichiometric. This can cause chemical compatibility problems with surrounding materials that are sensitive to oxygen contamination. Consequently, close control of stoichiometry is essential to acceptable irradiation behavior. However, since fission products consume less oxygen as oxides than is liberated during burnup of UO_2, oxide fuel becomes gradually more hyperstoichiometric as burnup proceeds.

Because of the low thermal conductivity of UO_2, temperatures and temperature gradients in most fuel designs tend to be high. Consequently, restructuring, swelling, and fission gas release figure prominently in the design of oxide fuels.

Fuel swelling is of particular concern in the design of the SP-100 reactor because of the potential for dimensional changes to disrupt the integrity of the heat pipes that cool the core. While the LANL staff had adopted swelling design curves based on available data (Buden and Stocky, 1982), others felt that the uncertainties were such that the swelling could be much larger. To resolve this issue, the Department of Energy convened a panel of independent experts. On the basis of a detailed review of the known data plus some significant new data uncovered by the SP-100 Fuel Panel, it was concluded that the design curves for UO_2 swelling adopted by LANL (Figure 6) were adequately conservative. It was also concluded that columnar grain growth would not begin below 1700°C, so that under the design conditions for SP-100 the vast majority of the fuel in the core would not undergo restructuring. However, the design of the SP-100, with unclad fuel plates, is particularly susceptible to swelling. The relatively large uncertainty in the swelling data does lead to difficulty in design, for instance, in how to provide for axial fuel expansion.

Some recent French (Commissariat a l'Energie Atomique (CEA)) data on swelling of UO_2 microspheres (Kauffmann et al., 1980) suggest a way to minimize swelling that could significantly simplify the design of a reactor such as SP-100. Figure 7 shows the CEA swelling data at

1500°C as a function of initial density. While the high-density material follows a fairly normal swelling curve, the low initial density material (92-95 percent) swells only a few percent and then saturates or even redensifies somewhat. Although the work done to date is somewhat sketchy, the key to this behavior appears to be the production of a controlled uniform fine porosity. With further development, this type of fuel could be very attractive for the SP-100.

Oxide fuel has also been proposed for use in the cermet form, consisting of UO_2 with additions of 20-40 percent of the refractory metals tungsten or molybdenum. The principal advantage of cermet fuel is its greatly improved thermal conductivity relative to the oxide, as shown in Figure 8 (Buden et al., 1979). It may also exhibit improved swelling behavior, but this is not confirmed. The principal disadvantage of cermet fuel is the reduced fuel density that leads to increased core sizes and weights.

Carbide Fuels

At least three types of uranium carbide fuels have been used in nuclear systems: the monocarbide UC, the dicarbide UC_2, and mixed carbide UC-ZrC. Mixed thorium-uranium carbide has been used in HTGRs such as the Fort St. Vrain Reactor. Each of these has its own unique characteristics.

Uranium monocarbide has a uranium density nearly 30 percent higher than that of UO_2 (Table 2) and a thermal conductivity nearly an order of magnitude higher (Figure 8). A major limitation in the use of carbide fuels in high-temperature applications is chemical compatibility with refractory metals that might be used as cladding or core structural materials. It can be seen from the free-energy curves in Figure 9 that niobium and tantalum will tend to reduce UC to the metal while forming refractory metal carbides. While molybdenum and tungsten are stable in contact with UC up to 2000 K or higher, Mo will react with any excess carbon to form a Mo_2C layer at the interface (Coen et al., 1969).

Uranium dicarbide was the reference fuel for the NERVA and Rover projects and is now the reference fuel for the rotating-bed and fixed-bed reactors. The Fort St. Vrain HTGR, now operating on the grid of Public Service Company of Colorado, is fueled with $(Th,U)C_2$. All of these systems have in common the use of pyrolytic carbon coatings. As can be seen from the phase diagram in Figure 10, the dicarbide phase is the equilibrium phase in contact with carbon at high temperatures. At low temperatures, U_2C_3 is the equilibrium phase. However, in practice, the UC_2 is nearly always stabilized by trace impurities and persists at all temperatures. As long as UC_2 is used in the form of coated particles with pyrolytic carbon coatings, it is an excellent candidate for high-temperature systems. Coated particles will be discussed in more detail in a subsequent section.

The mixed-carbide fuel 90% UC-10% ZrC was under development in the in-core thermionic program at General Atomic Company (Yang et al., 1972). To improve the dimensional stability and the thermionic performance of carbide-fueled emitters during in-pile operation, it was necessary to exercise rigid control of the structures, compositions, and stoichiometries of the carbide fuel material. To facilitate fission gas release in order to reduce fuel swelling, the carbide fuel bodies had low densities (75-79 percent theoretical value) and contained a large amount of open porosity. To prevent the dissolution of the tungsten cladding at the operating temperature of the emitter, about 4 wt% of tungsten was added to the carbide fuel material. The C/U atom ratio was set between 1.03 and 1.05 in order to reduce the diffusion rate of uranium through the tungsten cladding and to minimize the carburization of the tungsten cladding. Such carbide fuel material consisted of a uranium-zirconium carbide matrixcontaining a few weight percent of dissolved tungsten and dispersions of $U-WC_2$ phase. The fuel bodies usually contained a central hole of about 10 percent of the fuel cavity volume in order to accommodate fuel swelling. Irradiation testing showed that this structure was effective in reducing swelling below that of oxide fuels.

Nitride Fuels

Uranium nitride is the least studied and the least understood of the potential high-temperature fuels. However, nitride fuels have been considered both as a fuel for space power reactors and as an advanced fast breeder reactor fuel (Bauer, 1972). A good compilation of the state of the art in FBR fuel is given in the Proceedings of the International Conference on Fast Breeder Reactor Fuel Performance (Craig, 1979). Uranium nitride has the highest uranium density and the highest thermal conductivity of any of the unalloyed high-temperature fuels (Table 2). It also appears to have the lowest irradiation-induced swelling rate, as was already shown in Figure 3.

Uranium mononitride has good compatibility with most of the potential cladding materials, such as molybdenum, niobium, tantalum, vanadium, chromium, and iron. However, it will tend to react with the more reactive metals, such as zirconium, titanium, and aluminum. It is also compatible with liquid alkali metals.

A key concern with uranium nitride fuel is its high vapor pressure at high temperature, as illustrated in Figure 11 (Buden et al., 1979). The vapor pressure reaches 1 atm at about 2200 K. This is a particular concern for the SP-100, where the fuel is unclad and operates in a vacuum. Practical application of nitride fuels at high temperatures will require a nitrogen overpressure and some form of cladding.

COATED-PARTICLE FUELS

Coated-particle fuels have been developed both for the NERVA and Rover programs and for the commercial HTGR programs in the United States and Europe. The development history of coated-particle fuels has been summarized by Scott (1975), Simnad (1975), and Taub (1975). The status of coated-particle technology as of 1977 is covered in detail in a special issue of Nuclear Technology (Gulden and Nickel, 1977a).

A typical coated particle (Gulden and Nickel, 1977b) is of the order of 1 mm in diameter and consists of a central, fissile or fertile ceramic kernel surrounded by two or more coating layers. A commercial nuclear reactor will contain 10^{10}-10^{11} coated particles. Common to all current coated-particle designs (but not to the NERVA fuel) is the use of a porous, inner pyrolytic carbon buffer layer designed to accommodate kernel swelling, absorb fission recoils, and provide void volume for the accumulation of gaseous fission products. The outer coating layers serve as fission product barrier and pressure vessel, and generally consist of dense, isotropic pyrolytic carbon (Kaae et al., 1977) sometimes used in combination with chemically vapor-deposited silicon carbide (Price, 1977). The silicon carbide layer acts as a barrier to the release of metallic fission products and provides mechanical redundancy to the coatings. Zirconium carbide is an alternative barrier coating that has even better high-temperature performance. The various coating layers can be seen in the schematic and in the scanning electron microscope (SEM) micrograph of a deliberately cracked particle in Figures 12a and 12b. A wide choice of fissile kernel composition is possible in coated-particle fuels. The NERVA fuel was UC_2, which is compatible with the carbon coatings. Commercial and experimental HTGRs have used both oxide and carbide fuels. A considerable amount of development work has been performed on a two-phase, oxycarbide fuel that combines some of the advantages of both oxides and carbides (Homan et al., 1977).

For space power or other high-performance nuclear systems, some modification of current coated-particle technology may be desirable. Two key factors are materials compatibility and fuel loading. In a system such as the SP-100, pyrolytic carbon outer coatings are not acceptable, because of the potential for carburization and embrittlement of the Mo-Re alloy heat pipes. Coatings of zirconium carbide (Hollabaugh et al., 1977), tungsten, or some other refactory metal would be more compatible. In a more advanced nuclear system with a carbon or graphite core structure, more conventional coated particles with pyrolytic carbon coatings could be used to advantage, probably allowing higher temperature operation.

The other key factor in the use of coated-particle fuels in advanced nuclear systems is fuel loading. Fuel loadings are maximized both by minimizing coating thickness and by minimizing the interstitial void space in the bed of fuel particles. In general, fuel burnups in space power systems are low in relation to those in

commercial reactors. Consequently, coating thicknesses can be greatly reduced.

The random packing of spheres has been analyzed in detail (Scott, 1960; McGeary, 1961). While a randomly packed bed of uniform spheres will have a void fraction of 37-40 percent, a binary mixture of spheres with a size difference of at least seven will pack with less than 20 percent void fraction. With optimized coating thickness and optimized particle size distribution, the fuel loading for coated particles can approach that of the current SP-100 design.

The design of coated particles is complicated by the dimensional changes and creep of the carbon coatings. Neutron-induced contraction of the carbon coatings results in substantial shrinkage in the case of a two-layer (BISO) particle, or in compressive stresses in the carbide barrier coating in the case of multilayer (TRISO) particles. These effects are opposed by the internal pressure resulting from fission gas generation within the particle. Sophisticated coated-particle design models have been developed that consider these effects (Prados and Scott, 1966; Kaae, 1969; Bongartz, 1977). Owing to the complexity of the processes and the difficulty in obtaining precise input data, these models are still semiempirical. However, their value has been proven in coated-particle design and in the analysis of experimental results. Because of the inherent variability in coated-particle dimensions and properties, statistical methods are used in design (Gulden et al., 1972). This is illustrated in Figure 13.

Thermochemical processes can limit coated-particle-fuel performance under certain conditions. Two phenomena are of particular significance: (1) the so-called amoeba effect in which the fuel kernel tends to migrate up a temperature gradient through the carbon coatings, and (2) reactions between certain metallic fission products, notably palladium, and SiC coatings. It has been shown theoretically and verified experimentally that the amoeba effect in carbide fuels is controlled by solid-state thermal diffusion of carbon through the kernel under the influence of the temperature gradient (Gulden, 1975; Stansfield et al, 1975). This is illustrated in Figure 14. An amoeba effect is also observed in oxide fuels. Although its mechanism is not as well understood as that in carbide fuels, extensive experimental studies have shown that the amoeba effect can be treated phenomenologically the same as in carbide fuels (Wagner-Loffler, 1977). It has been shown that the presence of a zirconium carbide seal coating around the fuel kernel eliminates amoeba migration in coated-oxide fuel particles, and a similar effect is probable in coated-carbide fuel particles. Thus this particle failure mechanism can be eliminated with appropriate design changes.

Reactions between metallic fission products and the SiC coating have long been observed. However, only recently have sufficient experimental data been available to allow quantitative empirical design methods to be developed (Smith, 1979). Fission product reactions do not occur with zirconium carbide barrier coatings, and therefore zirconium carbide is preferable to silicon carbide as a

barrier material in coated-particle fuels that are intended for very
high temperature service where fission product reactions may cause
coating failure. Both of these classes of thermochemical phenomena
are sufficiently well understood to be treated quantitatively in
design. They are particularly important for high-temperature-fuel
applications.

Over the past 20 years, coated-particle fuels have been developed
to a high degree of sophistication and predictability. Potential fuel
improvements have been identified that may provide even greater
flexibility in their use in the future. The basis for the use of
coated-particle fuels in advanced nuclear systems is well established.

SUMMARY AND CONCLUSIONS

While a variety of high-temperature fuels is available for advanced
nuclear systems, the choice of a fuel type is controlled by a complex
interaction between system design constraints and fuel properties and
behavior. Key considerations include fuel compatibility with cladding
and coolant, fuel swelling and fission gas release, and density of
fissile material.

Uranium dioxide has become the standard fuel for most reactors
operating today that use conventional metal-clad fuel. It has
excellent compatibility with cladding materials and good irradiation
behavior, and perhaps most important, it has an impressive and
successful record of use. Monocarbide and mononitride fuels remain of
interest for advanced applications because of the substantial
improvements they offer in both fissile density and thermal
conductivity. However, materials compatibility issues must be
carefully considered with the carbides, and vaporization is a concern
with the nitrides. In both cases, but especially with nitrides, the
advantages for a particular system must be substantial to outweigh the
relative lack of experience and data as compared to the oxide.
Cermets, especially UO_2 with refractory metals, and the mixed
carbide UC-ZrC are also potential high-temperature fuel choices.

Coated-particle fuels were developed for the NERVA and Rover
programs and have proven very successful in experimental and
commercial high-temperature, gas-cooled reactors. Coated-particle
fuels are well developed, and the factors that control their
performance are well understood. Coated particles offer potential for
higher temperature operation than any other solid fuel. The coatings
are very effective in isolating the fuel from its surroundings, so
that swelling and chemical reactions can be controlled at a highly
localized level. Coated-particle fuels offer a high degree of
flexibility in choice of both coating and fuel materials and are
compatible with a wide range of advanced nuclear system designs.

The range of available high-temperature fuels for advanced nuclear
systems is broad. In general, no single fuel will be optimal for all
systems requirements. The fuel selection process involves a trade-off

between systems design and materials properties. In some cases, the fuel properties can be modified to better match systems requirements, as in cermets, alloyed fuels, or coated particles.

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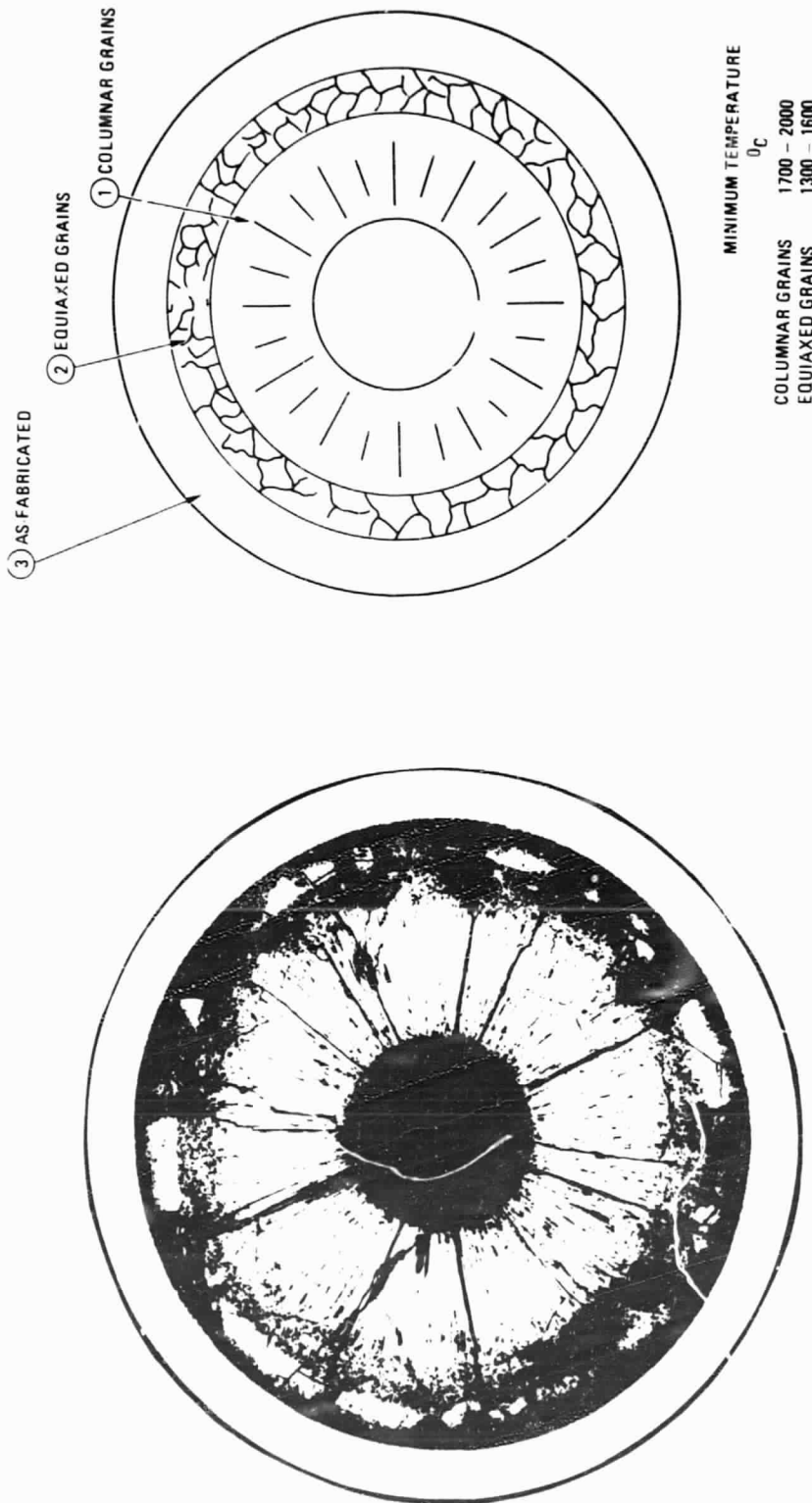
FIGURE 1 Fuel restructuring. (a) Cross section of mixed-oxide fuel rod irradiated to 2.7 percent burnup (O'Boyle et al., 1969). (b) Regions of a restructured fuel rod (Olander, 1976).
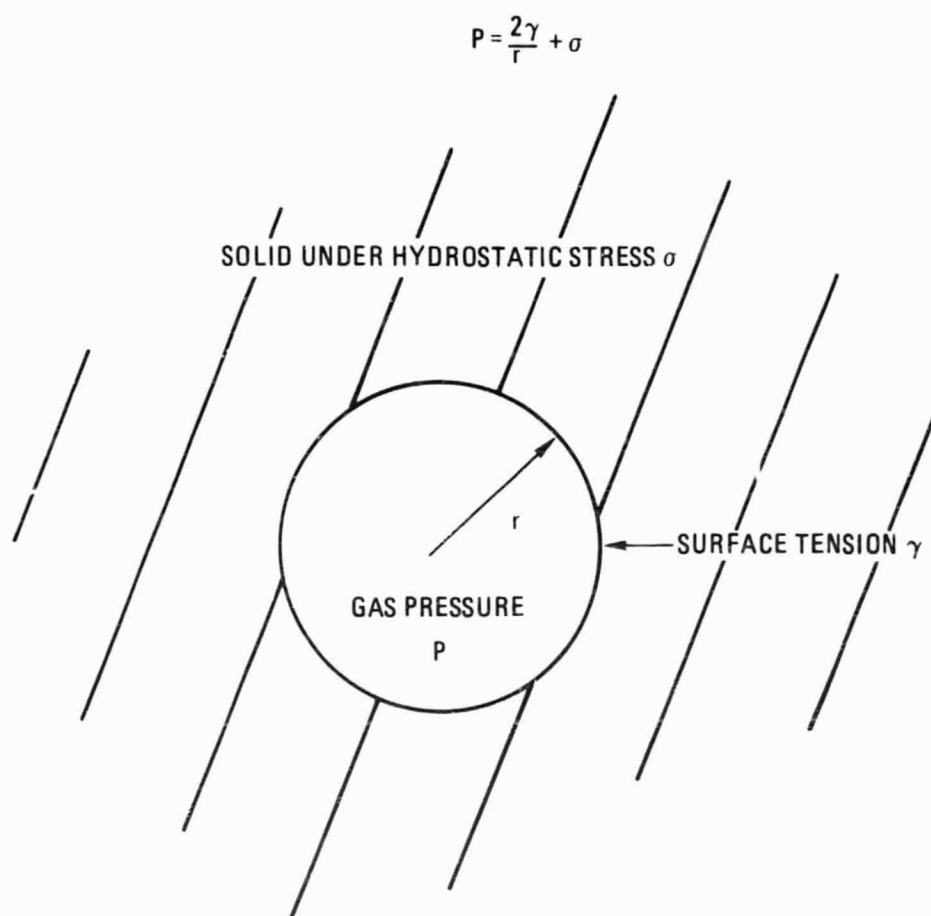
FIGURE 2 Gas-filled bubble in mechanical equilibrium with a solid under hydrostatic stress (Olander, 1976).
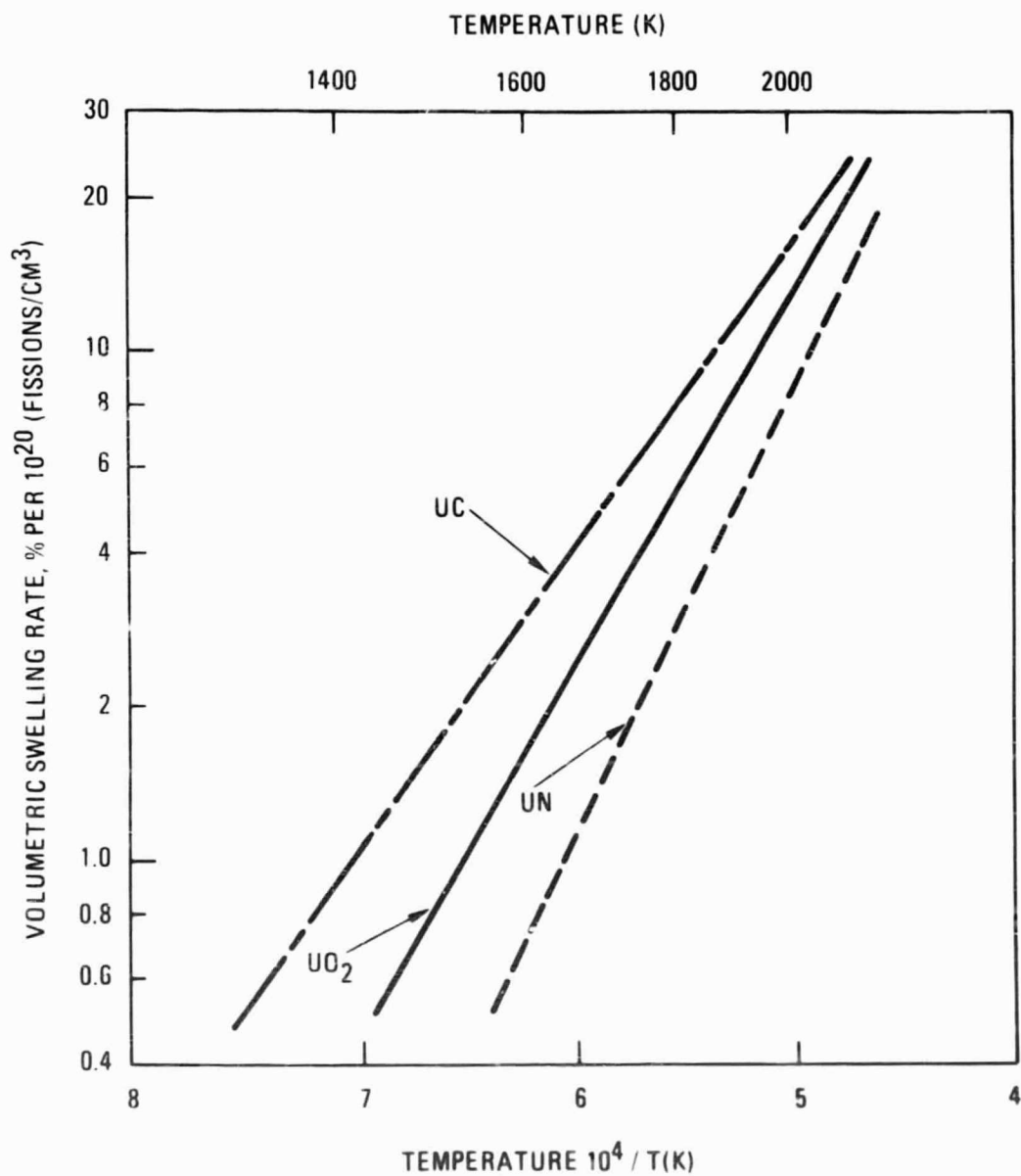
FIGURE 3 Dependence of fuel swelling on temperature for UC, UN, and UO$_2$ (Bauer, 1972).
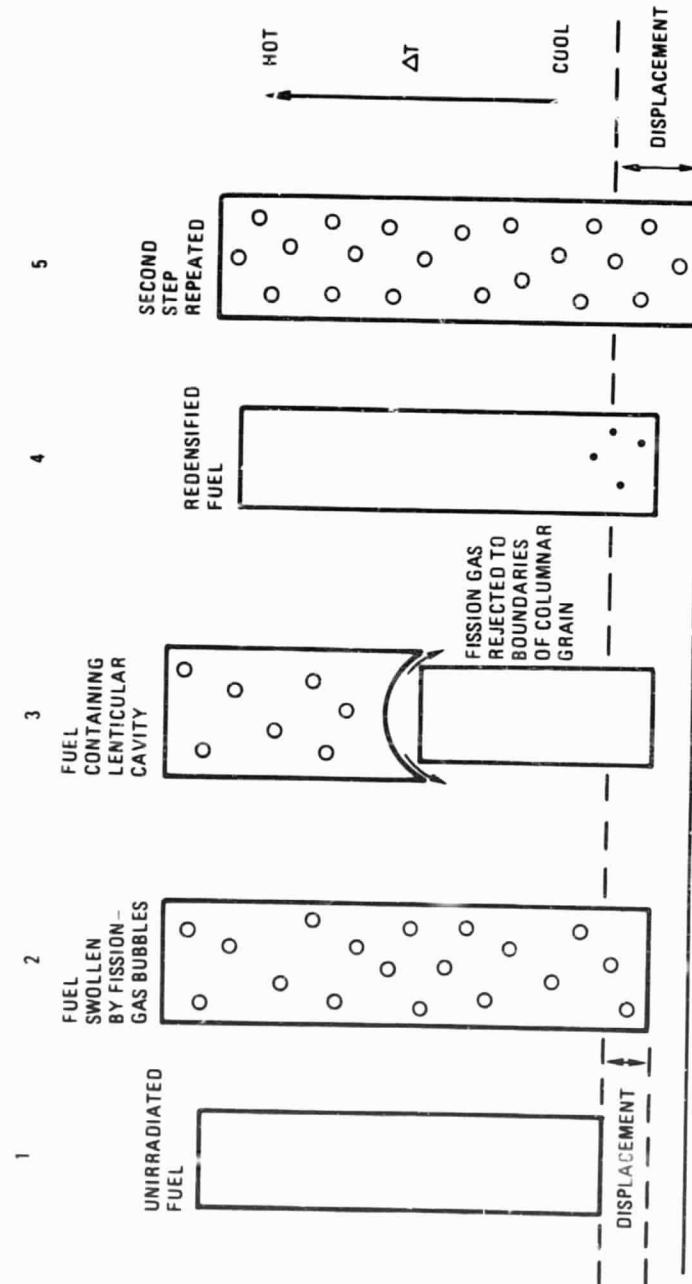
FIGURE 4 Schematic diagram of the swelling and gas release process due to swelling by lenticular cavities (Olander, 1976).
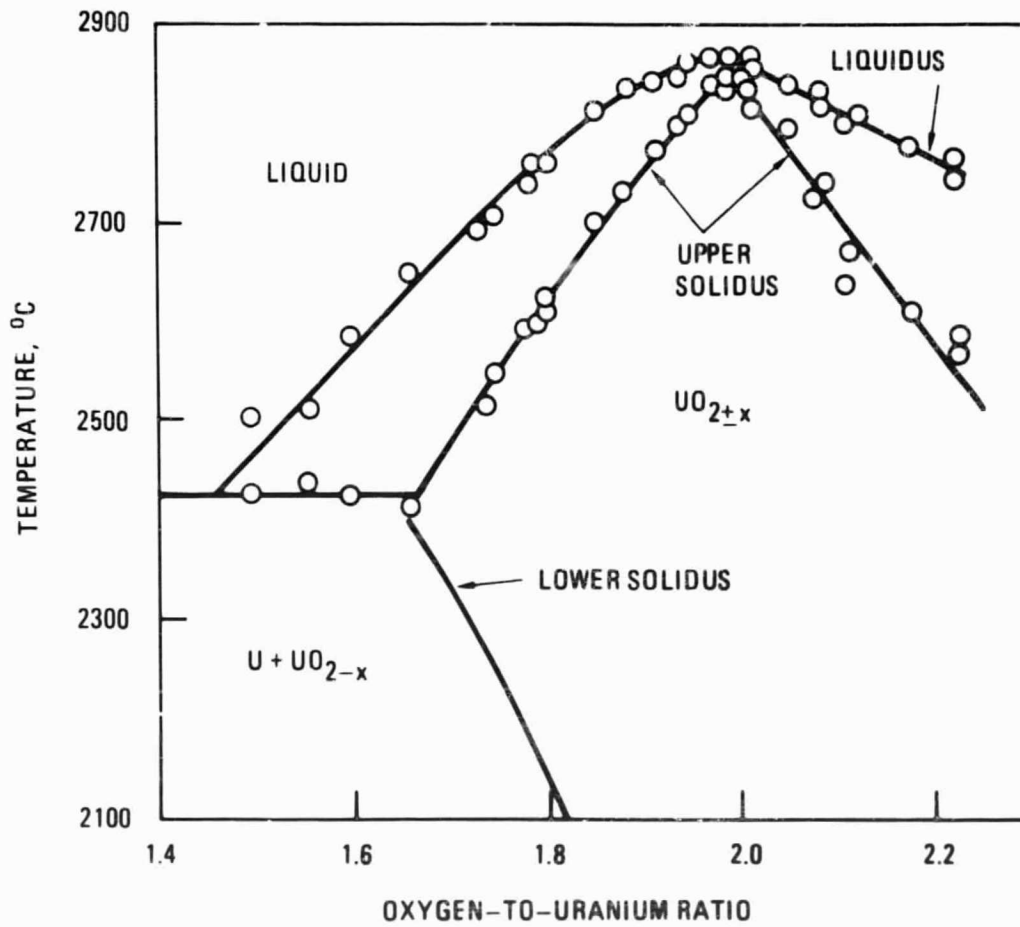
FIGURE 5 Partial phase diagram for urania from $UO_{1.5}$ to $UO_{2.23}$ (Latta and Fryxell, 1970).
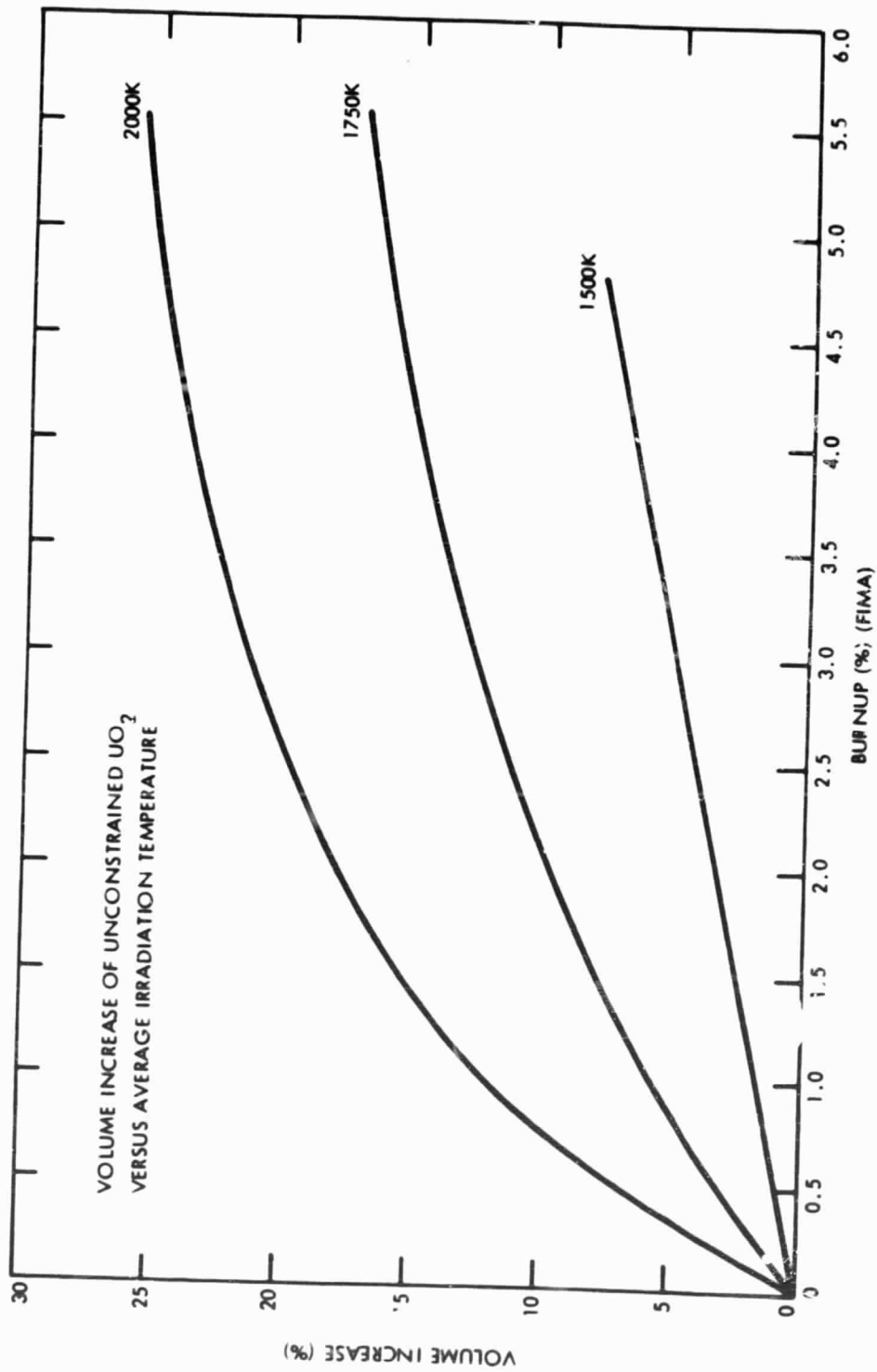
FIGURE 6 Fuel swelling curves for SP-100 core design (Buden and Stocky, 1982).
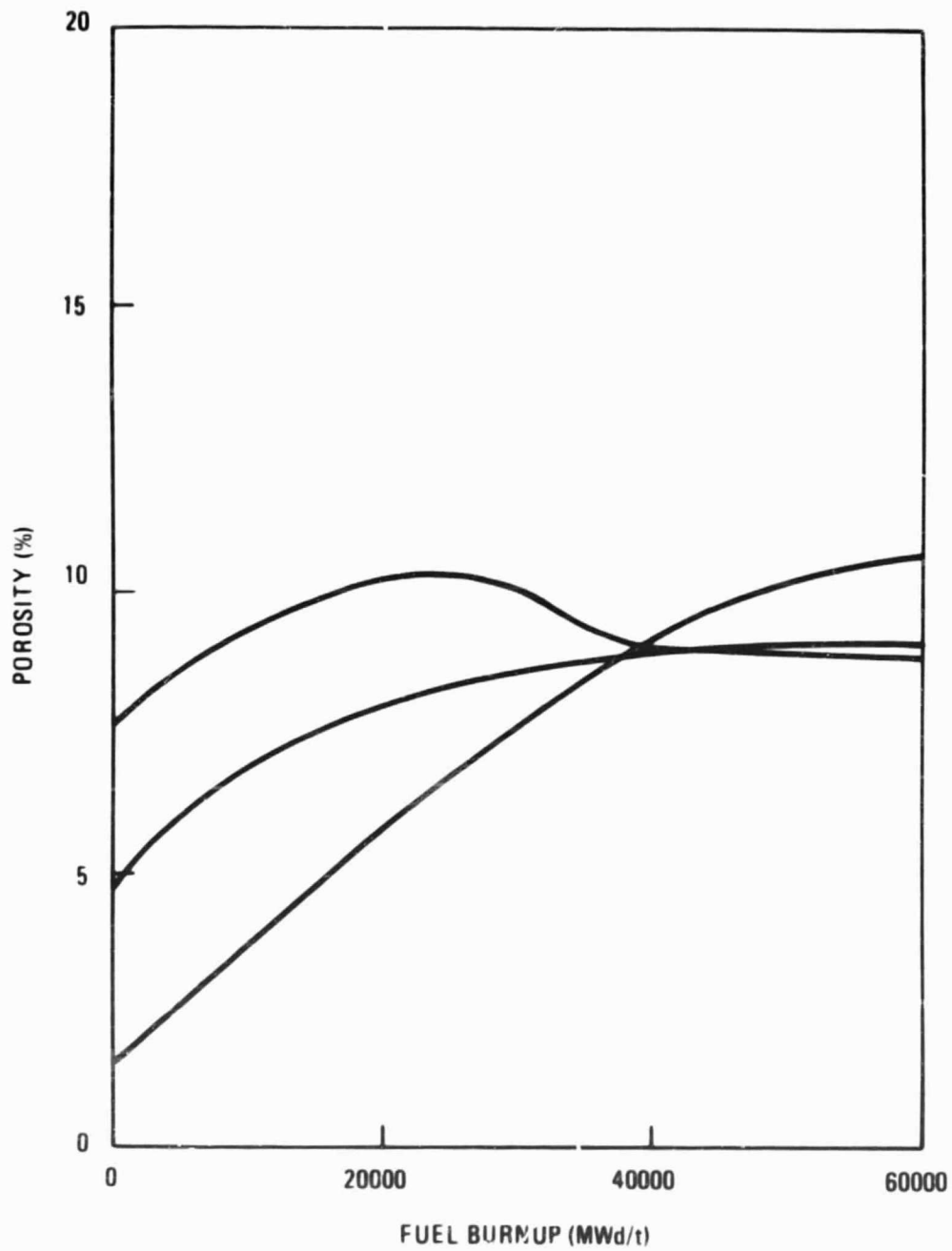
FIGURE 7 CEA data on swelling of UO_2 with various initial densities at 1500°C (Kauffmann, et al., 1980).
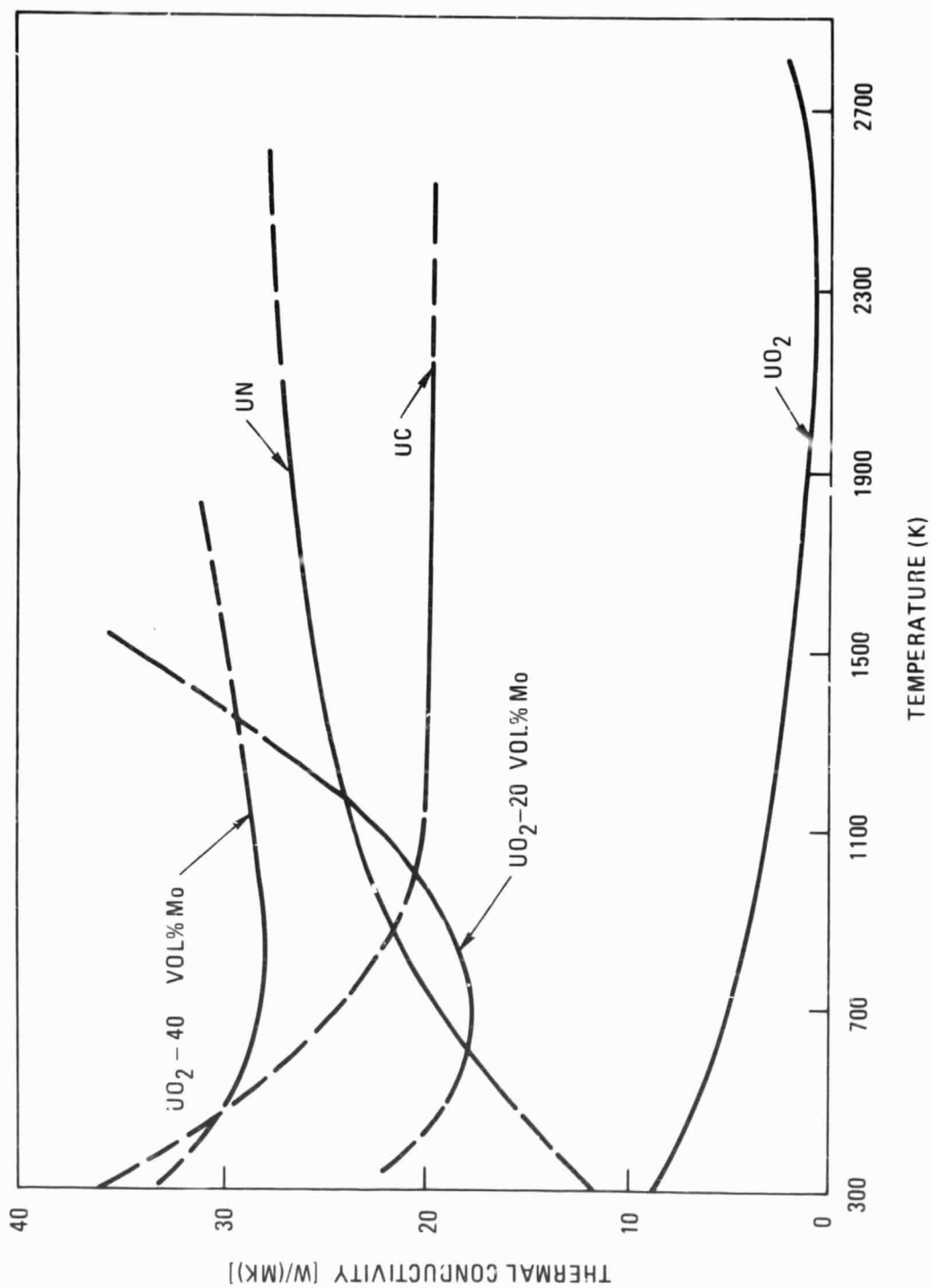
FIGURE 8 Thermal conductivity versus temperature for various fuels (Buden et al., 1979).
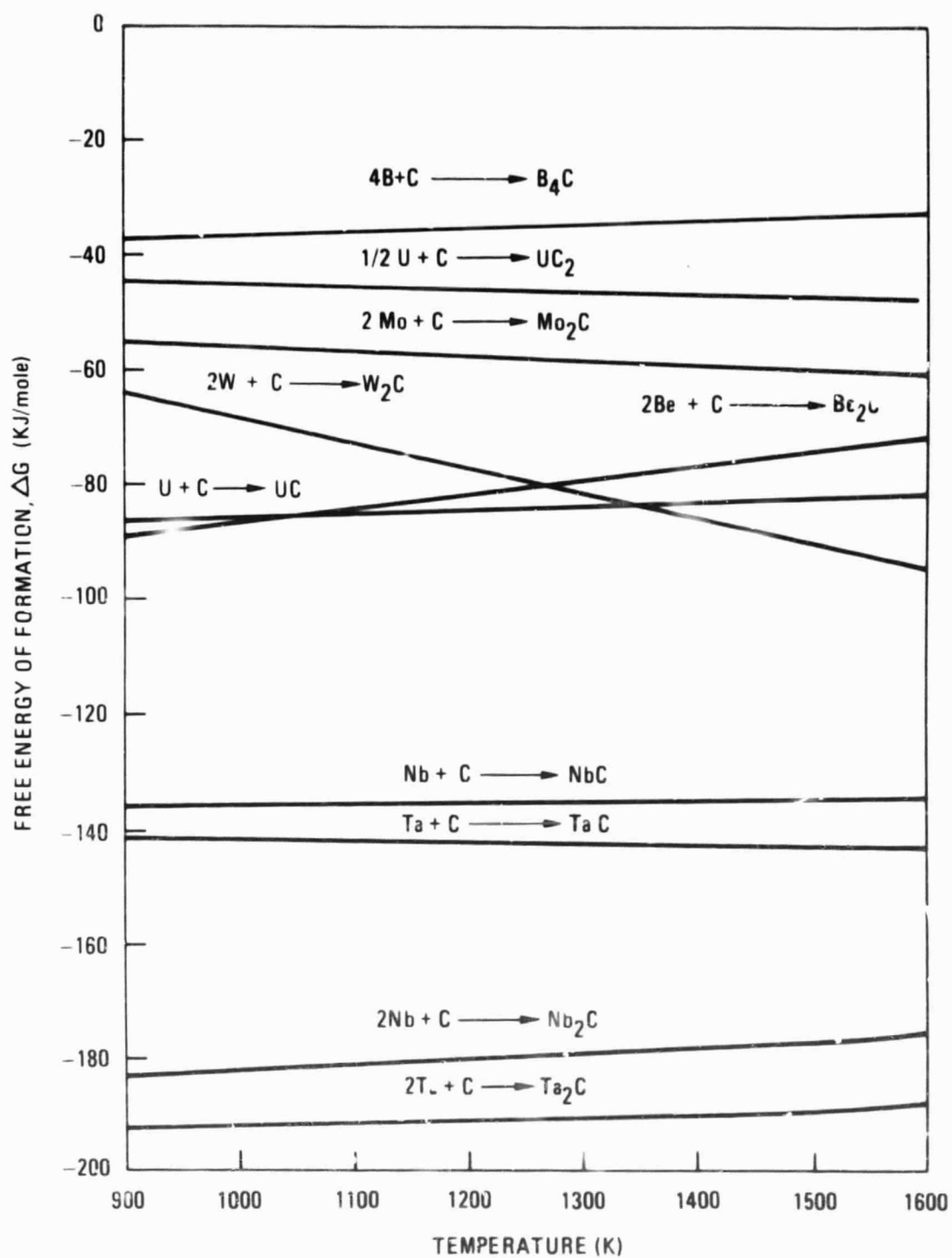
FIGURE 9   Comparison of free energy of formation (Coen et al., 1969).

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FIGURE 10 Phase diagram of the U-UC₂ system.
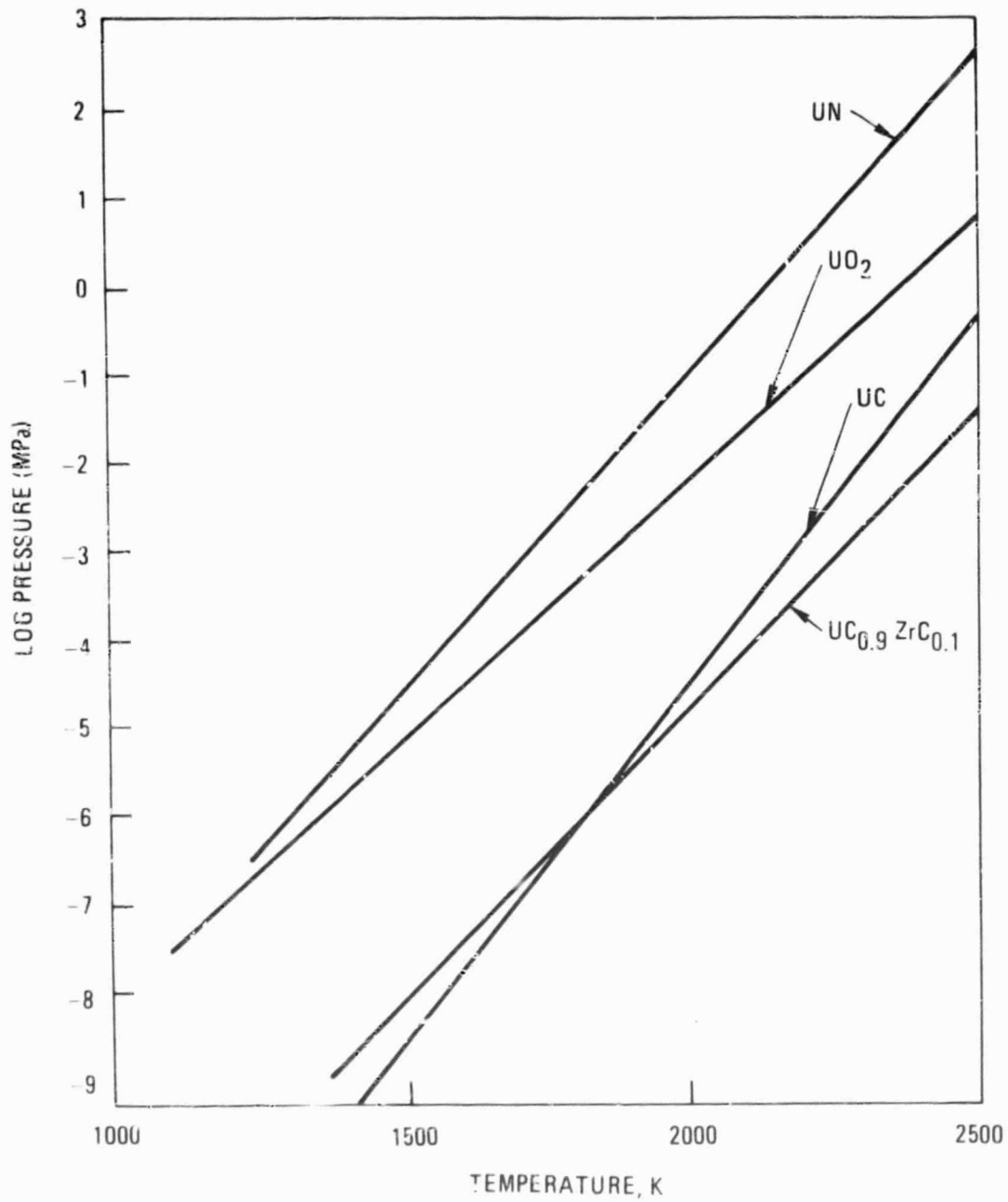
FIGURE 11   Vapor pressure of various fuels (Buden et al., 1979).
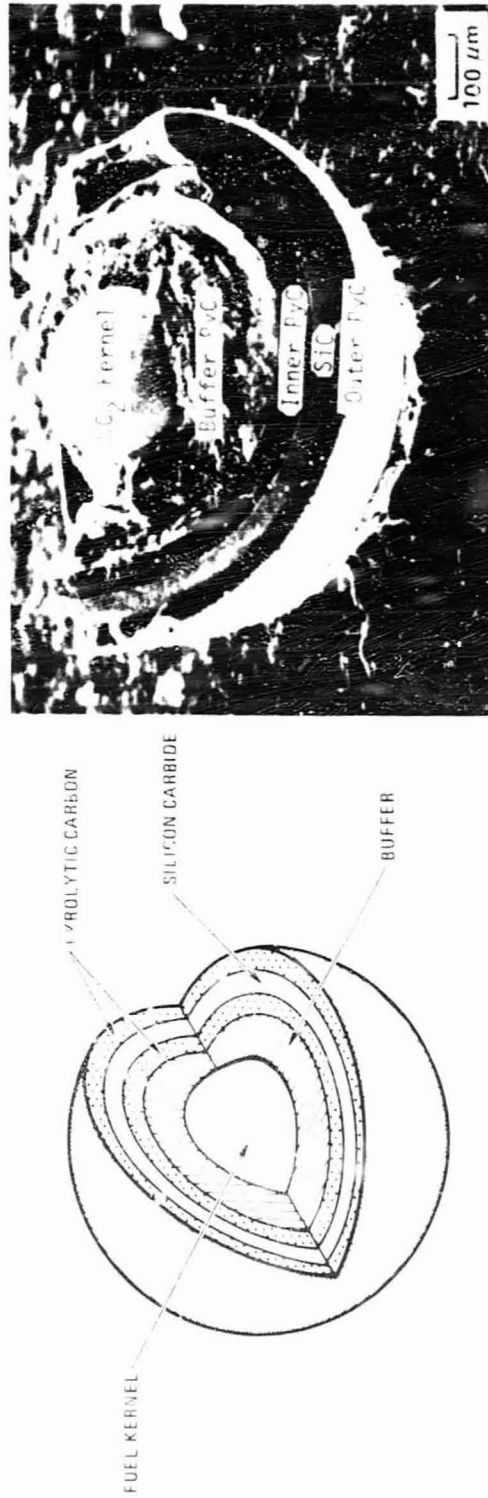
FIGURE 12 The coated-particle concept. (a) Schematic of TRISO-coated fuel particle. (b) SEM micrograph of a deliberately fractured TRISO-coated fuel particle.
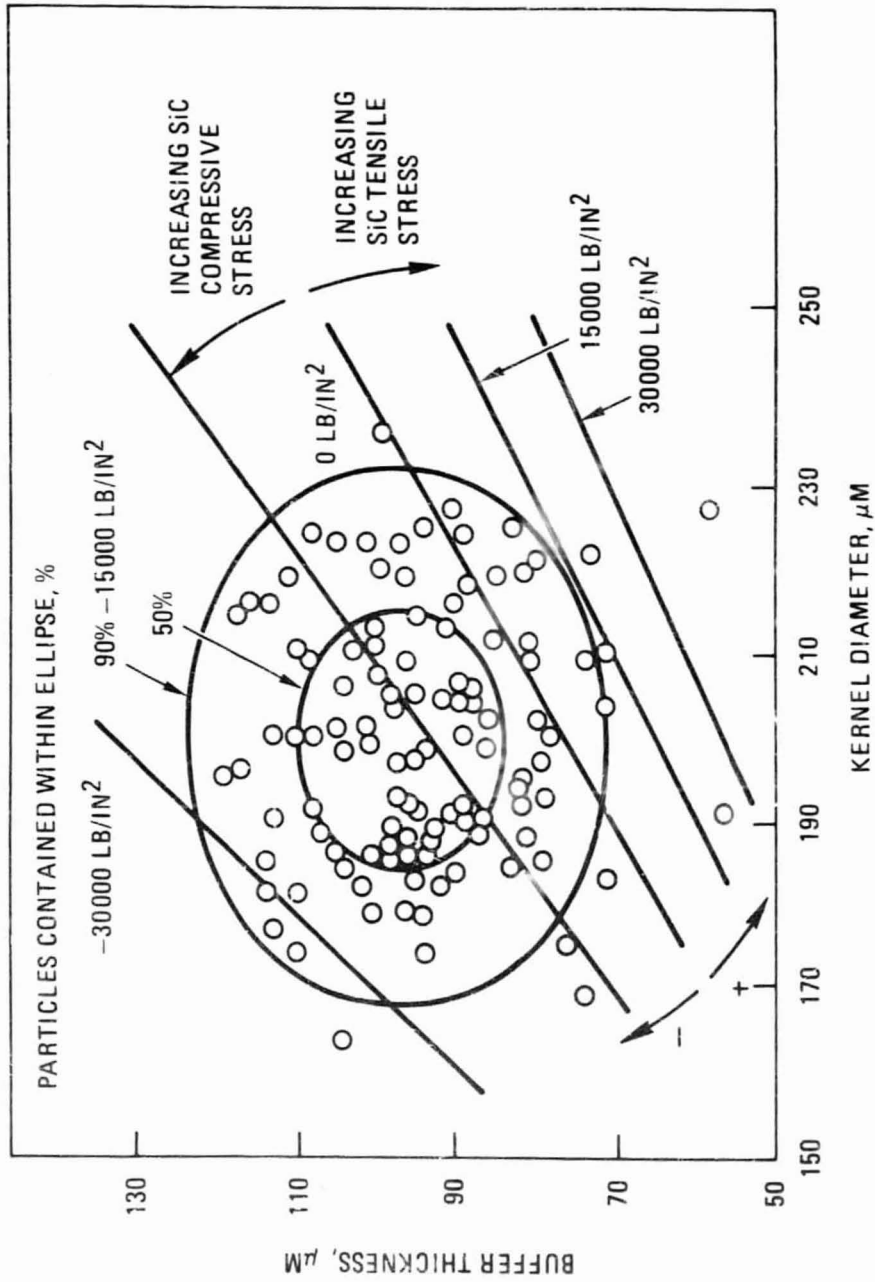
FIGURE 13 Two-dimensional distribution of a kernel diameter and buffer thickness for a TRISO-coated-particle batch with lines of constant end-of-life SiC stress.
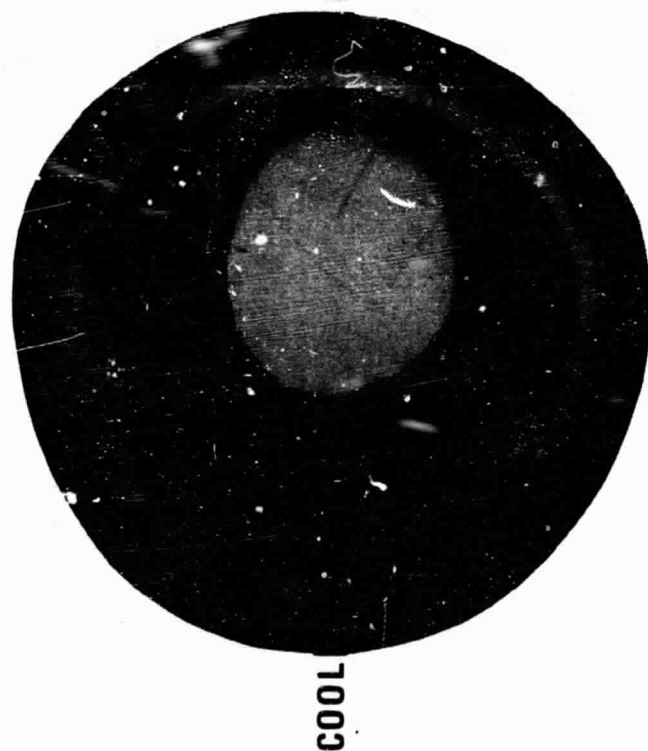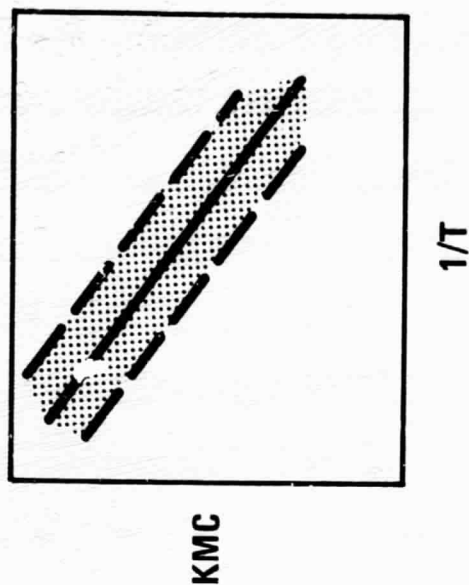
Fig. 14. The amoeba effect in a coated UC₂ particle. The fuel kernel has migrated to the hot side of the particle as a result of carbon diffusion through the kernel to the cool side. The kinetics of the process are described by the Kernel Migration Coefficient (KMC).

NEUTRON PHYSICS OF REACTORS FOR USE IN SPACE

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The power systems especially suited to serving applications in space could vary substantially in characteristics, depending on the mission. The range of important features might be even wider than is encountered in terrestrial power plants.

Some missions would require small to moderate amounts of electric power over long periods of time, perhaps a number of years. Some would require the ability to generate large amounts of power over short periods of time. For some of these, the bursts of power might be summoned several times in the life of the plant. For others, the need might arise only once. In designs for pulsed demand, the plant would probably be cycled at least once for test purposes and then would hibernate, working at low station-keeping power until called on to perform in earnest. The time available for response might be very short--a few seconds.

It is unlikely that any single design of power plant is best suited to spanning the full range of requirements. It is also unlikely that any single set of features, in terms of engineering or neutron physics, offers optimal performance over the range. The power plant should be tailored to the mission to which it is dedicated. Its thermal, thermal-hydraulic, and neutronic characteristics are the principal variables to be settled in design.

Although I will concentrate on the neutronic aspects of the programs that would be required for space reactors, the thermal and thermal-hydraulic aspects are intimately intertwined with the neutronic ones, and it is not possible to discuss them separately. Therefore I shall stray into these thermal and thermal-hydraulics areas but will not give a separate review of them

SYSTEMS CONSIDERED

We consider three classes of system among the many that have been proposed over the past 2-3 decades. These are of special interest at this time, for various reasons.

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The Rover/NERVA designs used fuel consisting of uranium carbide dispersed in graphite, and hydrogen as the coolant. This was a well-thermalized system. The weight of the reactor varied, but it is typified by the NERVA design of 3.4 tons for a reactor intended for a long-term (more than 60 min) burn at 1,500 MW.

The SP-100 is the most recent version of a series of designs that have been developed at Los Alamos National Laboratory over a period of years. It consists of a fast reactor core of UO_2 wafers sandwiched between sheets of molybdenum metal, which serves as a medium to conduct the heat to a system of heat pipes. The heat pipes transfer the heat to thermoelectric elements. Additional heat pipes transport the residual heat to a thermal radiator. The core is reflected on the sides and on one end by beryllium metal, and on the other end by beryllium oxide. This is the end that the heat pipes penetrate on their way to the thermoelectric junctions and the eventual radiative heat dissipation.

The third class of system is that based on use of particle beds. The particles are the coated spherical fuel particles developed for the High-Temperature Gas-Cooled Reactor (HTGR), and even earlier for the Rover/NERVA reactors. The bed takes the form of a cylindrical shell. The coolant is hydrogen forced through the bed from the outside surface and removed from the core region during exit at the inner surface. The Fixed-Bed Reactor core is a particle bed of this kind retained between two gas-permeable frits, with a reflector of beryllium on the outside and either beryllium or graphite on the inside. The Rotating-Bed Reactor would not have the inner frit. Rapid rotation of the reactor would press the bed outward against the frit, through centrifugal action, a tendency opposed by the inward moving coolant, which would fluidize the bed. There would be no inner reflector.

The SP-100 concept is directed to long-term generation of moderate amounts of electric power. The design point that has been most extensively explored has as an objective steady state electric power of 100 kW(e) generated from 1,400 kW(t), produced over a period of 7-10 years.

The Rover/NERVA designs were conceived of as propulsion power plants, to deliver typically about 55,000 lb of thrust from 1,500 MW of power. We assume in this paper that Rover/NERVA designs might be modified to generate electricity.

Particle bed reactors can be tailored to a wide range of requirements. Steady state power can be generated by a Fixed-Bed Reactor, driving electrical generators through (for instance) a Brayton-cycle gas turbine, producing from 100 kW(e) to a nominal 100 MW(e), assuming ability to dissipate the waste heat. Higher powers can be generated using a Rotating-Bed Reactor design. These power levels would be more easily achieved in bursts, when short-term methods of waste energy dissipation can be used.

NEUTRON PHYSICS REQUIRING EXPLORATION

It is necessary to develop a good physical understanding of the neutronic behavior of candidate systems for use in space applications. These systems all differ from the ones normally used for other purposes, such as for terrestrial power generation or for research, and so the theoretical methods used for design in the other areas will generally need further testing and refinement in these new applications. Good understanding is especially urgent because the power plants must be designed to operate over long periods with no repair, modification, or maintenance. The possibility of direct human intervention is discussed at times, but if this ever becomes possible in the future, it will still be a highly exceptional undertaking.

The literature contains detailed lists of calculated characteristics and features of designs based on the various reactor concepts that have been explored. These include neutronic features as well as engineering design parameters. The neutronic characteristics are based in each case on some particular scheme of computation in use at the time, usually one developed for other reactor systems. Very few of the computational schemes were benchmarked against assemblies resembling the ones explored for space applications. Although the design points are quite detailed in their description, the neutronic aspects are not nearly suited to reliable design. In many cases, important features of the computational schemes used to generate the tables of characteristics have been lost and cannot be reconstructed. The features include old neutronic machine codes, old cross section sets, recipes used for calculation or interpolation or extrapolation, assumptions as to configurations used in the calculations, etc. Any decision to use designs following a given reactor concept must recognize the need for an experimental neutron physics program that proceeds almost from a starting point where some basic neutron physics questions are settled--not because the calculations are meaningless but because they must be unusually reliable.

The neutron physics questions include those of long-term static performance. The initial reactivity of the system must be in the operating range contemplated at start-up and throughout system life at operating conditions. This requires close knowledge of the cold clean criticality, the reactivity effect of proceeding to temperatures suited to power generation, and the reactivity effects of burnup and fission product buildup over the life of the plant. For a typical plant, an uncertainty of 0.5 percent in reactivity could mean an uncertainty of 25 percent in system life. Additional details may be needed on such questions as the spatial variation of power and its change over plant life, radiation damage, and gamma-ray heating.

Systems that may be called on to deliver high power over short periods of time will require close knowledge of criticality; the effects of temperature, burnup, and fission product buildup on reactivity; the prompt neutron lifetime; the spatial variation of

power; the characteristic curves of control elements; and delay constants associated with reactivity feedback mechanisms and other possible sources of instability.

Safety considerations will require accurate knowledge of characteristics of the reactor that could be important during launch failures, such as reactivity effects of core distortion and core flooding. Safety considerations will also require knowledge of shutdown margins, including those associated with partial failure of control systems.

Some of these characteristics are not easily obtained in experimental programs but must be derived from calculations using reliable theory. It is therefore essential that the experimental programs include means of ensuring that the theoretical methods are dependable even when they cannot be tested. This is often managed through experiments that in themselves may have little direct relevance to operating properties, but that are important. An example is the experimental determination of activation rates and reactivity coefficients of various materials at different points in the reactor. These test the calculated features of the neutron spectrum, and thus indirectly they test the ability of the calculations to predict accurately features that depend on the spectrum.

CRITICALITY

The geometry of Rover/NERVA type cores is simple, since these cores would be nearly homogeneous. A number of these systems were built in both critical experiments and operating experiments, and criticality was explored. However, these studies were conducted about 20 years ago, at a time when the theoretical methods in vogue were much simpler and less exact than those used now, and fitting analysis to experiments required some manipulation of the calculations. If the details of the old experiments are still in existence and accessible, the work done with these assemblies can serve to benchmark calculations completed with more sophisticated theory and better cross sections than were available 20 years ago. I suspect that in first tries at calculating criticality of the old systems, a value of k_{eff} within about 0.5 percent of critical should not be thought of as shameful, and improving on this would take time and study.

The criticality of SP-100 is less well known. A series of rough mock-ups of just critical versions of a reference design was constructed in the Honeycomb assembly machine at Los Alamos, and it is reported that the calculated values of k_{eff} of the critical systems were very close to unity. This is encouraging, but it must be recognized that the process of experimental testing of neutronics has only begun in these studies, which were intended as the start of a substantial program. Better mock-ups are needed, particularly ones that distribute the components in a more realistic way, with simulated heat pipes providing paths for streaming of neutrons. These mock-ups

should also carefully reproduce the full effects of the beryllium reflector, as it provides a return current of thermalized neutrons and as it augments the effective value of neutrons-per-fission through (n,2n) and (δ,n) reactions.

The criticality of the particle bed system is even less well known at this time. Only one-dimensional transport calculations of k_{eff} have been done, but no criticality experiments. The analysis is still in the development phase.

TEMPERATURE EFFECTS

The effects of temperature on reactivity of reactors are produced in a number of ways. The causes include thermal expansion, resonance broadening from Doppler shift, altered equilibrium energy distribution of a thermal-neutron distribution, and phase changes, if they occur. The change in average energy of the thermal-neutron distribution can affect the balance between fission and parasitic capture, and it can alter the fraction of neutrons that leak from the reactor before inducing nuclear reactions. All the reactors we are considering here use enriched uranium fuel and have small Doppler coefficients.

The Rover/NERVA reactors had reasonably well thermalized spectra. The effects of temperature on reactivity were caused primarily by the increase in mean temperature of the thermal neutrons in thermal equilibrium with the moderator, as the temperature of the moderator was increased. The principal effect of this was an increase of neutron leakage with rising temperature. A second important effect of this change is an altered ratio of parasitic capture and ^{235}U fission, a ratio which is not constant in the low-energy region. Increased moderator temperature leads to increased mean thermal-neutron energy, which reduces reactivity through a greater reduction of fission than capture. In principle, these effects in Rover/NERVA cores should be well known, because the cores were operated both cold and hot. As was pointed out above, the theoretical methods 20 years ago were not so well developed, and the calculations did not fit the experimental results as well as might be desired. The ability of the old experiments to benchmark the new theoretical methods and cross sections depends on existence, accessibility, and completeness of the experimental record.

The particle bed reactors are also thermal systems, with most fissions induced by moderated neutrons returned from the reflector. The physical separation of moderator and reflector has an important bearing on the effect of temperature on reactivity. The outer reflector will have a temperature well below that of the exit coolant gas; it is cooled on one side by radiative heat losses into space, and on the other side by the cool inlet gas stream. The heat source in this medium is heat radiated from the hot fuel, and gamma-ray and neutron heating. This source is not very large. The temperature of the outer reflector will therefore not change rapidly with power, and

so the leakage of neutrons into space, which is governed by this temperature, will not be coupled closely to exit coolant gas temperature. Likewise, the mean energy of thermalized neutrons returning to the core will also be insensitive to the reactor power. Thus, the Rotating-Bed Reactor is expected to have a small temperature coefficient of reactivity and a small power coefficient. The inner reflector of the Fixed-Bed Reactor will operate at a temperature near that of the exit gas (higher if cooled only by exit gas, lower if there is partial cooling by a bypass stream of entry gas). The tendency of the temperature of the inner moderator to follow that of the exit gas contributes to the temperature and power coefficients through cross-section effects, but not through neutron leakage or thermal expansion. The net effect is expected to be small, but the sign is not known.

The SP-100 is a fast reactor, but thermal neutrons returned from the reflector play an important role in the criticality. Since the reflector is not actively cooled, its temperature should change as the temperature of the core changes. The reflector should contribute to the temperature and power coefficients through effects that resemble those appropriate to thermal reactors. These have their origin in neutron leakage and neutron energy dependence of competitive neutron absorption at thermal energies. They would add to other temperature effects from thermal expansion of the core and from altered distribution of the working fluid (sodium) in the heat pipes penetrating the core.

There are no experimental data on temperature and power coefficients of particle bed reactors or the SP-100.

SPATIAL POWER PEAKING

One feature common to the SP-100 and the particle bed reactor concepts is the rapid spatial variation of the neutron flux near the interface between the core and the reflector. Both the energy and the angular distributions are highly space dependent in this region.

In the beryllium reflector, the effects of variable spectrum are felt through the density of the fast-neutron-induced reaction $^9Be_4(n,2n)2^4He_2$. This reaction induces radiation damage in the reflector through the generation of helium nuclei that at low temperatures accumulate interstitially in the beryllium lattice and at higher temperatures migrate to nucleate and develop small bubbles of helium. This induces volumetric growth near the core-reflector interface, which distorts the beryllium into a shape convex against the reactor core. The effect has been seen in other reactors such as the Materials Testing   ctor (MTR), and can be averted through design if it appears to be serious, such as by keeping the temperature low enough to inhibit bubble formation. In any case, the spatial

distribution of the high-energy-neutron flux in the beryllium must be well understood.

In the fuel region, it is the returning current of well thermalized neutrons that requires attention. This is the origin of localized power peaking, which can be very sharp. The attenuation length of a thermal neutron in UO_2 is less than 1 mm, and so power peaking effects in the SP-100 should be highly local and may be very important. Proper calculation of the effect requires considerable detail in the thermal energy region, a feature not present in codes normally used to analyze fast reactors. This peaking, combined with the poor heat conductivity of the UO_2, may induce local temperatures in fuel that set a limit on reactor performance. The dynamics of heat pipes in these high-temperature regions also require attention. It is important to know at what power level burnout occurs in the heat pipes contemplated for SP-100.

In particle bed reactors, peaking factors from return of well thermalized neutrons must be well understood for thermal-hydraulic design. In the Fixed-Bed Reactor, which has an internal and an external reflector, power peaking will occur at both the inner and the outer surfaces of the cylindrical shell that comprises the bed. The coolant gas flows from the outside surface of the bed to the inside surface, being heated on the way, and so it is the peaking at the hotter inside surface that must be well determined if there is concern about excessive heating of the fuel. Power peaking sets a minimum flow rate of coolant to avoid burnout. The small value of ΔT between the coolant gas and the fuel particles seems to ensure freedom from problems in normal operation, where the coolant gas may be limited to about 1400 K and the fuel particles may be only some tens of degrees hotter. Problems could arise in some transient cases, such as start-up and shutdown, and accidental loss of coolant flow. We shall come back to this later.

The Rover/NERVA systems are free from severe power peaking problems.

LONG-TERM BEHAVIOR

The long-term reduction in reactivity as burnup proceeds would be important only for designs intended to produce power over extended periods. Both the SP-100 and the particle bed concepts are capable of design for very high fuel loadings and so could be tailored to suit the requirements. The SP-100 designs are likely to be less sensitive to fission product poisoning than the particle bed designs. The greater part of the SP-100 fissions is produced volumetrically in the core by fast neutrons, and fission products in this region compete for neutrons according to the relative values of macroscopic cross sections. In the particle bed concepts, fissions are all produced by neutrons returning from the reflectors. Because of the power peaking at the surface of the bed, fission products will build up preferentially in this region of the Fixed-Bed Reactor, where they

will generate a curtain through which the returning neutrons must pass. A similar effect will be produced at the outer surface of the SP-100, but its overall effect on reactivity will be less. Because of constant migration of particles through the bed, the Rotating-Bed Reactor would not experience the effect. There is some discussion of Fixed-Bed Reactor designs in which the bed is re xed at intervals to eliminate the problem. Long-term behavior in this detail has not yet been explored.

KINETICS AND START-UP AND SHUTDOWN

The prompt neutron lifetime of the SP-100 is of the order of 10^{-7} s, while that of particle bed systems and Rover/NERVA systems is of the order of 10^{-3}. These lifetimes are so short that neutronic behavior would not limit rapidity of response if rapidity is desired. The actual limits to speed of response are set by thermal and mechanical considerations.

Rover/NERVA systems are limited in rapidity of start-up and shutdown by the possibility of thermal shock to the core, where the fuel is distributed in a matrix of moderator clad by a refractory material. In the experiments conducted 2 decades ago, temperature ramps for the core were limited to 100°-150°F/s. At higher ramp rates, fuel damage was encountered.

Start-up and shutdown of particle bed systems would also have to be reviewed from the standpoint of thermal shock to structure. The fuel itself would not be prone to difficulties of this origin. But the frit at the inner surface of the fuel in the Fixed-Bed Reactor may be stressed during rapid power changes. The matrix of the frit would be volumetrically heated and cooled by the coolant stream, and this would avoid internal localized stressing. But heat transfer to any structural members would be slower. It may be necessary to devise a careful programming of balanced change of coolant flow and reactor power, to maintain the most sensitive components at a reasonably constant temperature and prevent thermal shock.

These considerations are important to use of particle bed reactors or Rover/NERVA reactors in rapid response modes. It might be necessary for some missions to ramp the power up from station-keeping levels to full power in a few seconds. The thermal shock to fuel limits the ability of Rover/NERVA to respond. Thermal shock may also have to be analyzed for particle bed systems, but it is not expected to set the lower limit to response time, as long as proper design is followed. The limit should be set by the time required to ramp up the coolant flow. This may even involve switching from closed-cycle to open-cycle cooling.

The SP-100 is not at present considered for pulsed operation. Limiting ramp rates are still important, however, for reasons of safety and stability. At the operating power level of 1,400 kW, the adiabatic rate of core heatup would be about 200°C/s. It is

unlikely that the heat pipes could respond fast enough for cooling to track heating at these power levels, and so questions of stability must be considered. This point will be considered again in the following section.

SAFETY AND STABILITY

In the present context, safety means freedom from disabling failures. Considerations of safety and stability are very different for different space reactor concepts, and in fact, the most important features in this context are likely to be set by such details of design as pump and valve reliability, electrical system characteristics, etc. There are, however, some generic aspects of SP-100 and particle bed designs that will require exploration in a neutronics program. These arise from coupled neutronics, thermal performance, and thermal-hydraulics.

The delay times associated with heat transport in SP-100 could be important during power changes. These delays are the result of low heat conductivity of the uranium oxide fuel, and mass transport in the heat pipes. On the other hand, SP-100 is not intended for transient use but is designed to operate at a single power level over long periods of time. Instabilities from delayed feedback coefficients will be avoided if power changes are made at rates slow in comparison with the delay times. This may be important if any load-following operation is required.

A second generic question concerning SP-100 concerns the margins to thermal failure. Points that must be well understood are the margins to fuel failure at the hot spot, which is expected to occur at the core-reflector interface, where high flux peaking may be found, and the margin to burnout in the highest power heat pipe. In this connection, it will be necessary to explore probabilities and effects of control system malfunctions, such as might result during changes in system demand.

Two possible sources of instability of particle bed concepts must be investigated. One is the potential for channeling of the coolant gas through the bed of fuel particles. Another is any possible tendency toward instability of the meniscus of the inner fuel surface in the Rotating-Bed Reactor.

The generic safety issue pertinent to particle bed reactors operated in the pulsed mode with sharp power increases and decreases is, How are cooling capability and reactor power to be ramped simultaneously so as to match power generation with cooling? The question of highest interest in this connection is that of sensitivity of temperature of critical components to departures from optimum strategy.

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NEUTRONIC PROGRAMS

There was an extensive experimental program on KIWI/Rover/NERVA
reactors in the late 1950s and early 1960s, when these systems were
being developed for rocket propulsion. In the course of that program,
a number of systems were built, and a broad physics program was
pursued. This may provide an adequate basis for tailoring similar
reactor systems to more modern applications. However, to be certain
that no new experimental neutron physics is needed, it will be
necessary to review the archival information from these programs, to
determine its completeness and adequacy as a basis for benchmarking
calculations made using current cross sections and computational
methods. Without such a review, it is difficult to estimate how much
new experimental work would be needed. In any event, a substantial
theoretical program would be needed to redo the old theoretical
interpretations and to establish an analytical basis for confident
design calculations. Of course, any decision to review the
Rover/NERVA program would have to point toward a working demonstration
plant where the final neutron physics design could be checked.

Several delayed critical mock-ups of SP-100 have been studied in
the Honeycomb facility at Los Alamos. This experiment was, however,
only of an introductory character, primarily to provide a first test
of the ability to calculate criticality. These assemblies were also
used to conduct some safety tests and measurements of reactivity
coefficients of several materials in the core. These experiments were
a useful introductory set that established confidence in a number of
aspects of the theoretical methods. Additional experiments would
certainly be needed in a serious development program. These should
include more exact mock-up of designs contemplated, including
simulated heat pipes, detailed simulation of control drums, careful
reproduction of details of the core-reflector interface where accurate
knowledge of power peaking may be necessary, and a core size,
composition, and geometry close to the design point. This will, of
course, require ability to hold down, by use of distributed poison,
the excess reactivity designed in for temperature defect and burnup.

The experimental program should include determination of
criticality, reactivity coefficients of distributed poison for
extrapolation to the k_{eff} of the unpoisoned core, reactivity
coefficients of coolant in the heat pipes, neutron lifetime, power
distribution in the core (with high detail near power spikes),
high-energy-flux distribution in the reflector, control
characteristics of the control drums, safety experiments considering
launch accidents, and (to the extent possible) separate temperature
coefficients of the reflector and the core. It is unlikely that exact
temperature coefficients could be measured in any experimental
configuration short of a detailed demonstration experiment, but some
components of the coefficient can be explored with critical assemblies.

No neutronic experiments have been performed on particle bed
reactors. The exploratory analysis that has been done for these

concepts has been based on use of the one-dimensional transport code ANISN, which has been employed for many years and has been checked against a great many experimental assemblies, but not against any closely resembling the particle bed designs. A serious development program would certainly require an experimental neutronics program, directed to the same broad class of questions listed above in support of the SP-100 program. There would, however, be some differences, particularly the need to determine sensitivity of reactivity to disturbances in the particle bed. The experimental program could be started with approximate mock-ups as with SP-100, would then move to more exact mock-ups, and would continue into physics testing of a demonstration unit or units.

SPACE REACTOR SHIELDING: AN ASSESSMENT OF THE TECHNOLOGY

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ABSTRACT

Analytical and experimental techniques for designing and testing space
reactor shields were developed at Oak Ridge National Laboratory more
than a decade ago. The analytical techniques, which are available for
application to current systems, automatically optimize the size,
weight, and shape of the shield by careful placement of selected
shield materials around the reactor. The design of the shield varies
with angle from the reactor center, depending on the dose-rate
constraints specified for different points outside the system. The
selection of the materials used in the design is aided by experimental
determinations of the radiation-attenuating characteristics of various
candidate materials at the ORNL Tower Shielding Facility (TSF), which
uses a 1-MW(t) reactor as the source (the Tower Shielding Reactor II,
or TSR-II). Measurements with the TSR-II also confirm basic data
(such as cross sections) needed as input in the analytical methods.
Prototypic shield designs can also be tested at the TSF with a
10-kW(t) SNAP reactor used as the source. These same techniques can
serve as a basis for developing a space reactor shielding technology
to meet the demands of future systems. Such a shielding technology
program should include generic studies of various shield materials to
determine not only their shielding characteristics, but also their
material and structural integrity in severe temperature and radiation
environments and under launch and accident conditions. It should also
include generic studies of radiation streaming through penetrations in
shields such as those required to accommodate coolant flow and reactor
control. Finally, it should include experimental tests and detailed
analyses of prototypic shields. Only with such a technology will
designers be assured that the shield weight, which can dominate the
overall weight of space systems, is kept to a minimum while at the
same time being assured that the dose rates and total doses delivered
to the payload, to the reactor electronic components, and to the

maintenance and operating personnel do not exceed the maximum
allowable limits.

INTRODUCTION

All space power reactor systems must include radiation shields to
protect the payload, the components of the system itself, maintenance
personnel, and mission crews. Since the weight of the shield may
dominate the overall system design, the choice of the shield materials
to provide this protection and the optimization of the designed shield
with respect to size and shape are very important.

Shield materials are selected both for their radiation attenuation
characteristics and for their structural stability under severe
temperature and radiation environments and under launch and accident
conditions. But the shields are always compromised to so... degree by
penetrations required to accommodate coolant flow and control
devices. These penetrations provide paths through which neutrons can
stream and complicate the design. Thus, the final shield design for a
specific system must be based on a detailed shield analysis, which in
turn must be based on a sound space reactor shielding technology.
This paper discusses the requirements for space reactor shields, the
current capability for meeting those requirements, and the need for
additional studies to develop a space reactor shielding technology for
current and future systems.

REQUIREMENTS FOR SPACE REACTOR SHIELDS

Shields for space power systems must meet three specific radiation
design constraints. First, they must ensure that the radiation
exposure levels per unit time specified for the structural and
electronic components of both the payload and the reactor are not
exceeded. second, they must ensure that the integral radiation
exposures (total lifetime exposures) set for these components are not
exceeded. And, third, they must protect personnel. For unmanned
systems, it may be necessary to allow for the possiblity of reactor
and payload maintenance and docking operations. For manned
spacecraft, the design must ensure that the mission crew does not
receive dose rates or total doses above the specified limits.

The shield must also meet several structural design requirements.
Minimum weight is a primary consideration. In most space power
plants, either the shield or the heat rejection system dominates
overall system weight. Normally, the shields tend to dominate the
weights of spacecraft utilizing low power reactors, while the heat
rejection systems (i.e., radiators) tend to dominate the weights of
those utilizing higher powered units. Minimum shield size usually is
also important. In the past, considerable emphasis has been given to
optimizing the shape of a shield to reduce both its size and weight,

and such optimization will become even more important in the future,
especially if the systems are to be launched with a space shuttle.
The shield must also possess the material stability to withstand the
temperature and radiation environments required for routine
operation. And finally, it must possess the structural stability to
withstand launch conditions and to retain its integrity during
accident conditions.

EARLIER SPACE REACTOR SHIELD PROGRAMS

Several investigations of space reactor shielding designs have been
carried out at ORNL in the past. The most recent study, which ended
about 10 years ago, was primarily oriented to the Systems for Nuclear
Auxiliary Power (SNAP) program and included fundamental measurements
of the shielding effectiveness of some candidate shielding materials.
In addition, calculation methods and the physical data that they
require (such as cross sections) were developed to create an analytic
capability. These tools were used in designing several shield
concepts during this program.

The measurements, made at the ORNL Tower Shielding Facility,
utilized a modified SNAP-10A reactor (Mynatt, 1971). They included
basic radiation attenuation measurements for lithium hydride, ^{238}U,
lead, tungsten, and Hevimet (a tungsten-copper-nickel alloy), as well
as a few combinations of lithium hydride and high-Z materials. The
energy distributions of neutrons and gamma rays leaking from some
candidate shield materials were also measured and compared with
calculations.

Lithium hydride and tungsten were ultimately selected as shield
materials in the SNAP program, lithium hydride because of its light
weight and good neutron attenuation characteristics and tungsten
because it is a good gamma-ray shield and was believed to have a low
production of secondary gamma rays due to neutron capture. (It should
be noted that while these choices were rational for the SNAP program,
they are not necessarily the best choices for the current shielding of
space power reactor systems. Lithium hydride can experience hydrogen
loss due to the effects of temperature and radiation, and tungsten is
heavy and may not really represent a minimum secondary gamma-ray
source.)

In the SNAP shield designs, the lithium hydride and tungsten were
often arranged in laminated layers for increased efficiency. The
analytical techniques used to carry out the designs included
one-dimensional discrete ordinates radiation transport calculations
that optimized the thickness (weight) of the individual shield layers
and also their placement within the shields. The techniques also
included two-dimensional discrete ordinates calculations to "shape"
the shield layers and configurations, plus coupled two-dimensional
discrete ordinates and three-dimensional Monte Carlo calculations to
investigate effects such a radiation streaming through pipes and

crevices in the shields. These same techniques were also applied in several other shielding studies for space power systems (Engle et al., 1971a,b).

CURRENT ANALYTIC CAPABILITY FOR SPACE REACTOR SHIELD DESIGN

The analytic capability developed for the earlier programs is applicable to current space power reactor shield programs. Several examples follow to illustrate how the various available methods have been used--in these cases to optimize and shape the bottom and side shields for a 450-kW(t) space reactor (Engle et al., 1971b).

Bottom Shield Optimization

The one-dimensional optimization techniques, which are simple and have resulted in 30- to 50-percent weight savings in earlier applications, were used to determine the required thicknesses and locations of lithium hydride and tungsten layers in a two-cycle bottom shield (the shield between the core and the payload). The results are shown in Figure 1, where the thicknesses and locations of the successive shield layers are indicated for various dose-rate constraints. The inner surface of the shield begins at a radius of 18 cm.

For a dose-rate constraint of 10-mrem/h at 200 ft from the reactor center, the first tungsten layer of the bottom shield would have to be about 5 cm thick, the first lithium hydride layer approximately 20 cm thick, the second tungsten layer about 8 cm thick, and the second lithium hydride layer slightly over 40 cm thick. The overall shield thickness would be about 79 cm. If a more stringent dose-rate requirement of 1 mrem/h were mandated, the first tungsten layer would have to be increased to about 14 cm and the second tungsten layer to approximately 10 cm. The first lithium hydride layer would remain at the same thickness as for the 10-mrem/h dose constraint level, but the second lithium hydride layer would have to be increased by about 5 cm. In this case the total shield thickness would be about 94 cm. The difference in weights of the two shields would be proportionately greater than the differences in their total thicknesses owing to the large increases in the heavy tungsten layers. Diagrams such as Figure 1 can be constructed for multicycle shields and then used in conjunction with specific design constraints as a basis for initial shield configurations.

Side Shield Optimization

Figure 2 gives similar optimization results for a single-cycle side shield of tungsten and lithium hydride. Note that here the dose-rate constraint levels extend from 1 rem/h up to 100 rem/h at a distance of

100 ft from the reactor center. (Since the side shield does not
protect the payload, the constraints for the side shield are normally
less stringent than those for the bottom shield.)

For the single-cycle side shield configuration to meet the 1-rem/h
constraint requires approximately 45 cm of tungsten and about 60 cm of
lithium hydride. If, however, a 10-rem/h dose-rate constraint were
specified, only 5 cm of tungsten and about 52 cm of lithium hydride
would be required. A further relaxation of the dose-rate constraint
to 20 rem/h would allow the tungsten layer to be eliminated
altogether, and increasing the constraint level from 20 to 100 rem/h
would decrease the remaining lithium hydride layer from about 50 cm to
approximately 28 cm. Shield layer decreases such as those illustrated
here, and even the presence or absence of a given layer, are
determined automatically by the optimization code as it strives to
meet the dose-rate constraint.

Shield Shaping

The next step was to shape the layered shields around the reactor.
This was done by calculating isodose contours with a two-dimensional
discrete ordinates transport code. In a multilayered shield, the
neutron shield material outer boundary is shaped to follow the neutron
isodose contour, and the gamma-ray shield material is shaped to follow
the gamma-ray isodose contour. The dose rates resulting from the
revised shield configuration are then calculated to check the results
of the shielding shaping. This technique, which has resulted in 10-
to 20-percent weight savings, is illustrated in Figures 3-6.

Figure 3 shows a preliminary asymmetric reactor-shield
configuration, including the heat-pipe region located directly above
the core. In this case, the spacecraft is to be manned, and the
dose-rate constraint for the bottom shield (between the reactor and
the payload) is a factor of 100 lower than that for the top and side
shields, which are required only for approaches and docking
maneuvers. The bottom and side shields consist of a three-cycle
system of tungsten and lithium hydride, but toward the top the shield
decreases to a two-cycle system.

Figure 4 shows the neutron isodose contours calculated for this
preliminary shield. Note that the dose levels at the top are
approximately 10^5 mrem/h, while those near the bottom are 10^3
mrem/h, meeting the requirement of a factor-of-100 difference.
Shaping was done along the 10^3-mrem/h contour line for the outer
lithium hydride shield in the bottom area.

Figure 5 shows the corresponding isodose contours for gamma rays.
Note that here the isodose lines are very close together within the
tungsten shield layers. The outer tungsten layer was shaped at about
the 10^5-mrem/h contour line for the vertical portion of the shield
directly opposite the core.

The final shape-optimized shield configuration is shown in Figure 6. The savings in weight for this shield compared to that of the shield shown in Figure 3 is approximately 10 percent.

CURRENT CAPABILITY FOR DESIGN-CONFIRMATION EXPERIMENTS

Integral experiments are required to verify the analytical methods and data used for shield design and also to validate the final prototypic shield designs. Such experiments have been performed for many shielding programs, including the earlier SNAP shielding programs, at the ORNL Tower Shielding Facility. The TSF, shown in Figure 7, has a large experimental area and is located at a remote site, so that large and complex experimental shield configurations can be accommodated. The four large towers shown in Figure 7 were originally constructed so that a reactor could be hoisted into the air to provide a radiation source for measurements related to the Aircraft Nuclear Propulsion (ANP) program. The reactor now sets on a large concrete pad beneath the near towers and is enclosed by a concrete structure. The staff for this facility has many years of experience in performing shielding experiments, and the instrumentation used for shielding measurements has been well validated through experience. The experiments are designed to exploit state-of-the-art shielding analysis capabilities.

Two different reactors are available at the TST. The Tower Shielding Reactor II (TSR-II) is used for measurements to determine the adequacy of cross section data used in calculations of neutron and gamma-ray transport and secondary gamma-ray production. Experiments with the TSR-II can also be used to verify techniques for calculating radiation streaming through shield penetrations, as well as shield weight and shape optimization techniques. The second reactor, a modified SNAP reactor, is used to verify shield designs.

The Tower Shielding Reactor II (TSR-II)

The TSR-II is a spherically shaped reactor that operates at powers up to 1 MW(t). The high thermal power of this reactor allows neutron measurements of up to eight orders of magnitude reduction in fluence levels, which should cover the range of interest for space power applications.

Figure 8 shows the configuration of a shield that has been measured with the TSR-II as the radiation source. This particular test was for a gas-cooled fast reactor, and the first few layers of iron, aluminum, and boral comprised a spectrum modifier (to modify the spectrum of neutrons leaking from the TSR-II to simulate the spectrum of neutrons that would emerge from the core of a fast reactor). The uranium oxide following the spectrum modifer simulated the radial blanket for the gas-cooled reactor, and the graphite, boronated graphite, and stainless steel sections represented a prospective shielding

configuration. This figure indicates the deep neutron penetration in large dose attenuation type of measurements that can be performed with the TSR-II. It also indicates the modular natures of these experiments; each piece of the shield can be added individually, thus allowing measurements to be made at intermediate positions throughout the blanket and shield for comparision with calculations.

Modified SNAP Reactor

The modified SNAP reactor available at the TSF is basically a SNAP-10A reactor with a SNAP-10A reflector assembly and a SNAP-2 shield (see Figures 9 and 10). This reactor is rated for 10 kW(t) and uses highly enriched uranium in zirconium hydride fuel. It has a beryllium reflector with four control drum cutouts and uses NaK coolant. The top plenum has fin tubes to provide increased cooling. The horizontal section through the core (Figure 10) shows the coarse and fine control drums on the four corners of the reactor housing surrounding the central core.

The reactor's lithium hydride shield contains internal struts to provide increased stability, and the switches and wiring are rated for high-temperature operation. As indicated in Figure 9, other experimental shields (presumably prototype shields for specific reactor applications) could be substituted for the lithium hydride shield. This reactor has had a very low burnup, so that most of the core's lifetime is still available.

PROPOSED SPACE REACTOR SHIELDING PROGRAMS

It is important that further development of a shielding technology for space power systems begin with a generic shielding program. Of course, as designs are conceived, a technology for prototype shields would also develop.

Generic Shielding Program

The generic program would examine a set of verified methods and data for application to various design projects. Prospective shielding materials would be examined for their shielding effectiveness, both singly and in laminated configurations, and they would be classified by size and weight considerations, as well as availability and cost. Promising materials would be tested for their stability when exposed to high temperatures and high radiation levels. They would also be tested, in appropriate structural configurations, for shock resistance to simulated launch conditions.

The generic program would also include a study of geometric shielding effects. Radiation streaming through penetrations in the

shield would be examined as a function of the sizes and number of penetrations and of shield thickness. In addition, reflection from "out-of-shadow components" would be examined to determine the effect of placing structural members outside the shield shadow on the dose received by the payload or crew behind the shield.

Prototypic Shielding Programs

A prototypic shielding technology program is needed for each project. The shield composition and configuration must be selected to meet project-specific requirements, and the shield design must be analyzed and optimized in order to meet project dose-rate constraints. Where feasible, radiation fluence levels and energy distributions should be measured for each prototypic shield configuration. Analysis will be used to verify fluence levels in important positions in the shield configuration where measurements cannot be taken. The amount of effort required for a specific project will depend on the amount of work done in the generic program and on prototypic work done for earlier projects.

SUMMARY

Space power reactor systems require shielding to protect payload and reactor shielding components, and also maintenance and operating personnel. Shield composition, size, and shape are important design considerations since the shield can dominate the overall weight of the system. Techniques for space reactor shield design analysis and optimization and experimental test facilities are available for design verification. With these tools, a shielding technology in support of current and future space power reactor systems can be developed. Efforts in this direction should begin with a generic shielding program to provide information on materials properties and geometric effects and should be followed by project-specific shielding programs to provide design optimization and prototype shield verification.

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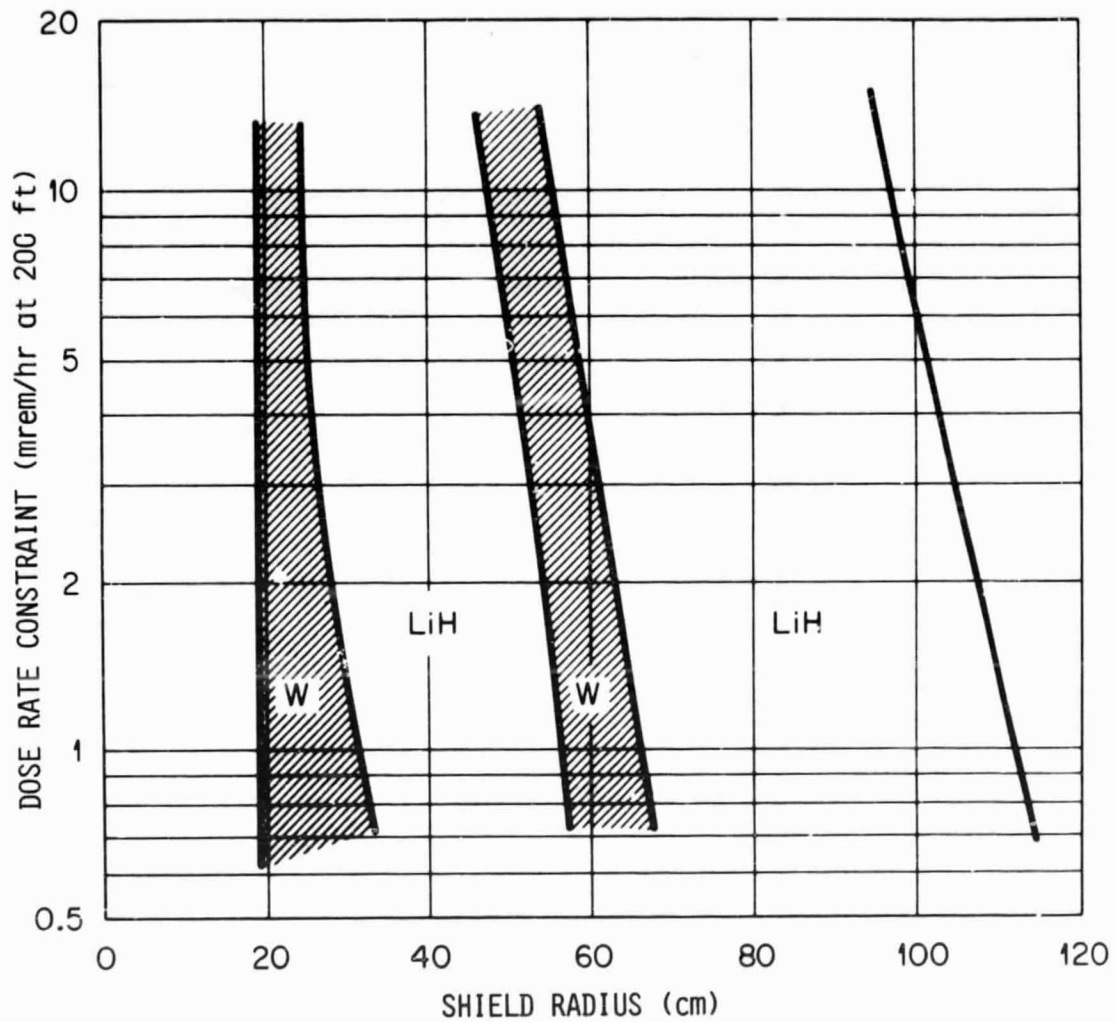
FIGURE 1 Minimum thicknesses of tungsten and lithium hydride layers in a two-cycle shield as a function of the dose rate at a distance of 200 ft from a 450-kW(t) reactor.
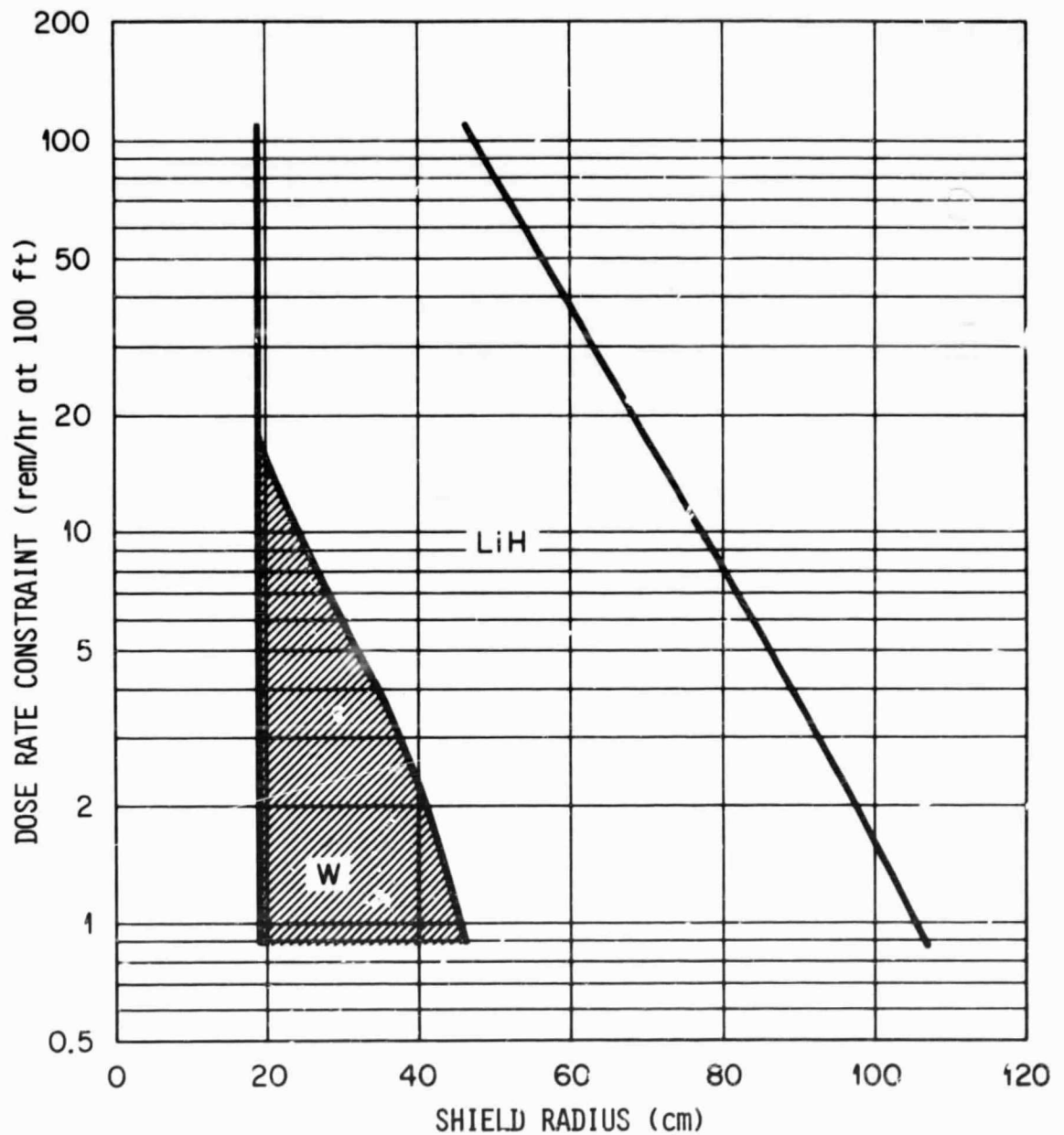
FIGURE 2 Minimum thicknesses of tungsten and lithium hydride layers in a one-cycle shield as a function of the dose rate at a distance of 100 ft from a 450-kW(t) reactor.
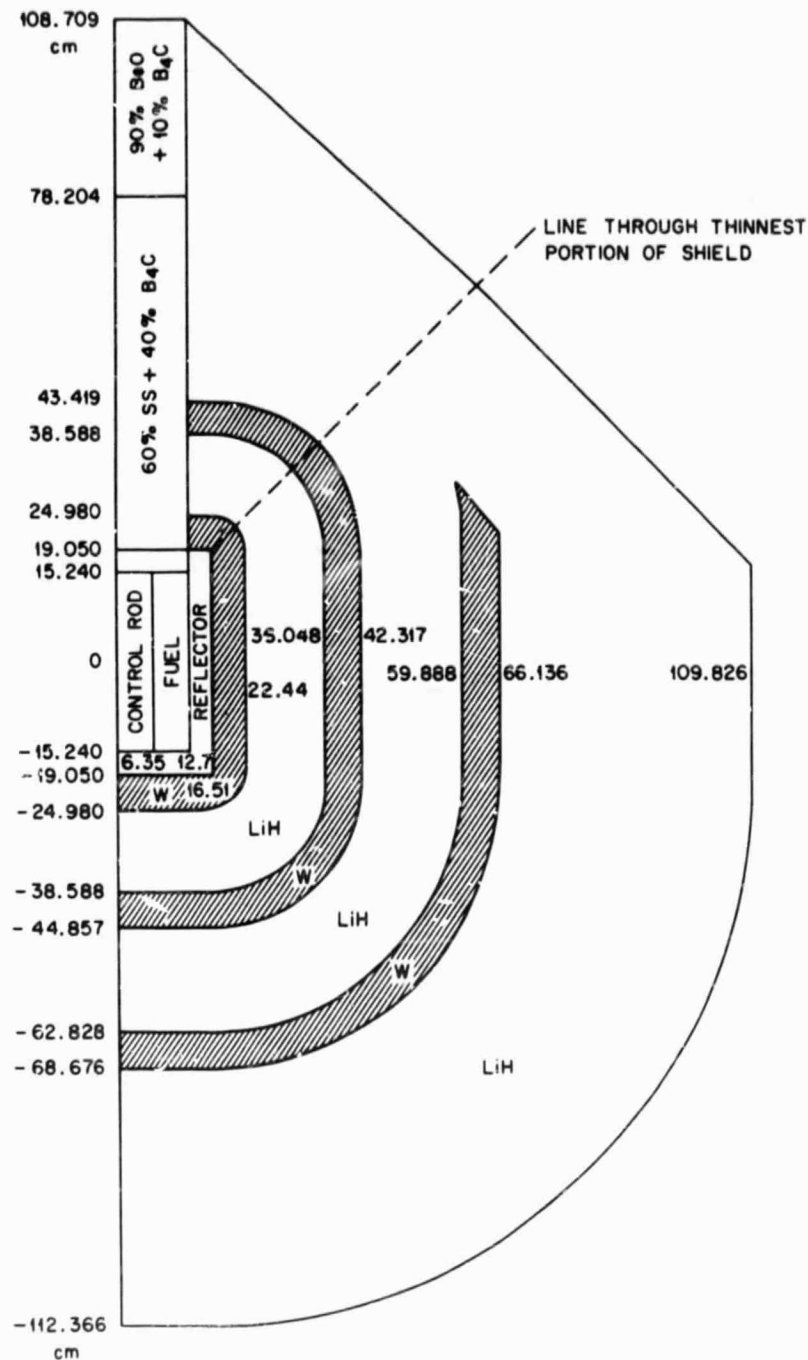
FIGURE 3 Preliminary configuration for an asymmetric space power reactor shield with a 90-degree cone angle.
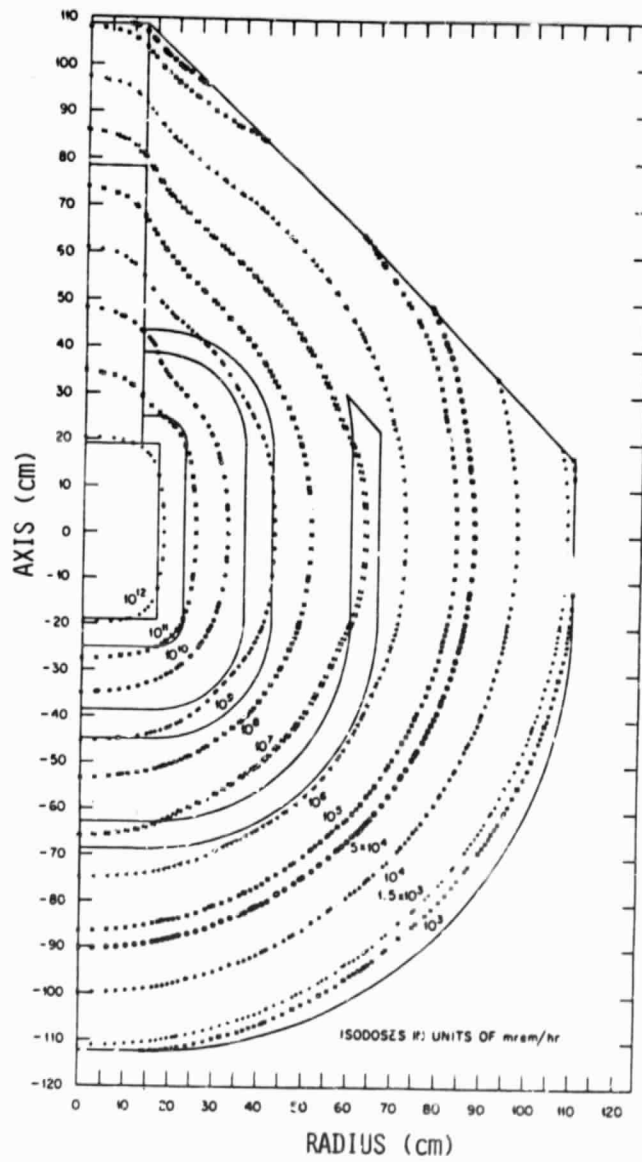
FIGURE 4 Neutron isodose contour plot for a preliminary asymmetric shield configuration.
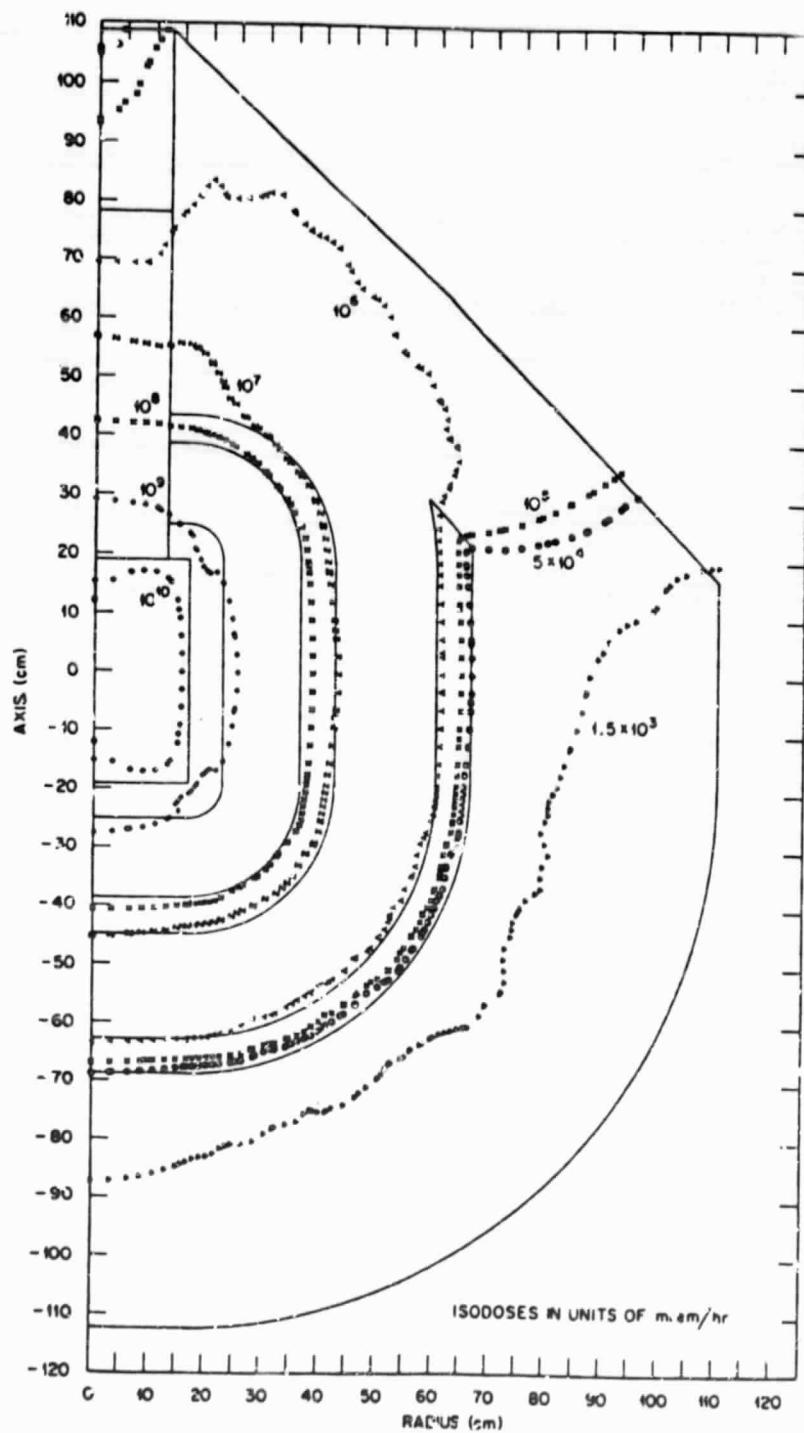
FIGURE 5 Gamma-ray isodose contour plot for a preliminary asymmetric shield configuration.
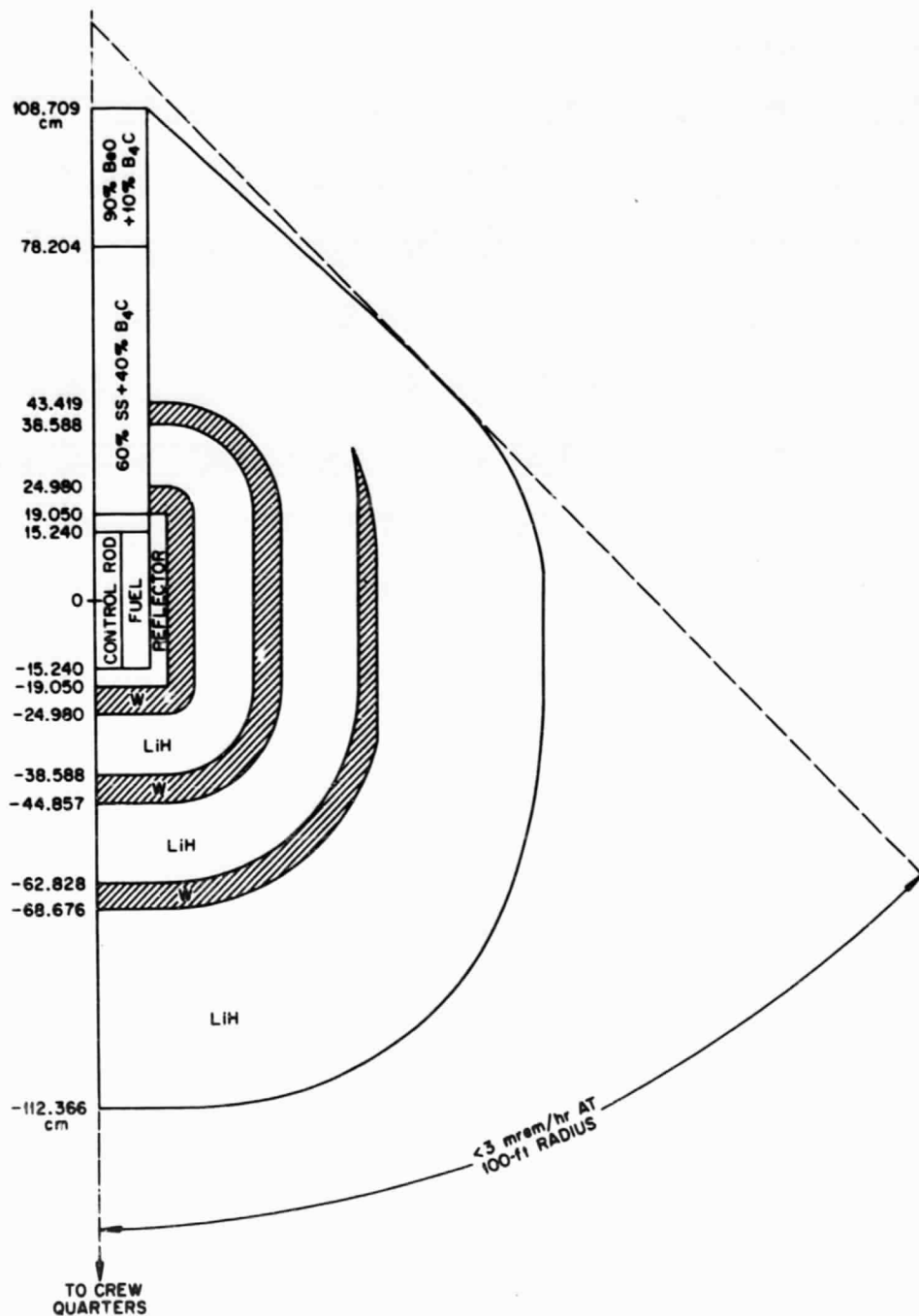
FIGURE 6 Shape-optimized shield configuration based on neutron and gamma-ray isodose contours.

FIGURE 7 The Tower Shielding Facility at Oak Ridge National
Laboratory.
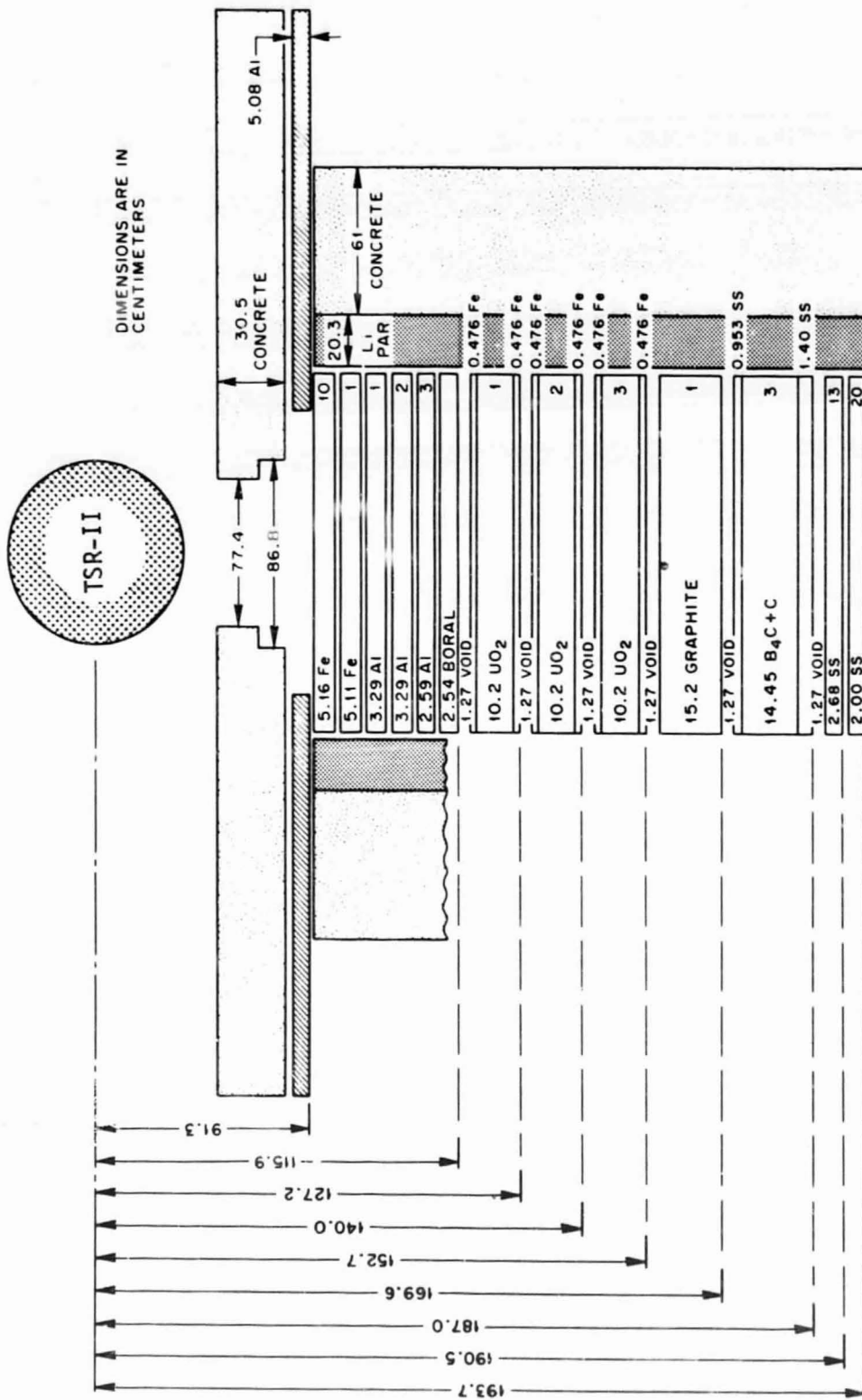
FIGURE 8 A typical TSF experimental configuration showing the TSR-II, beam port, and slab shielding experiments.
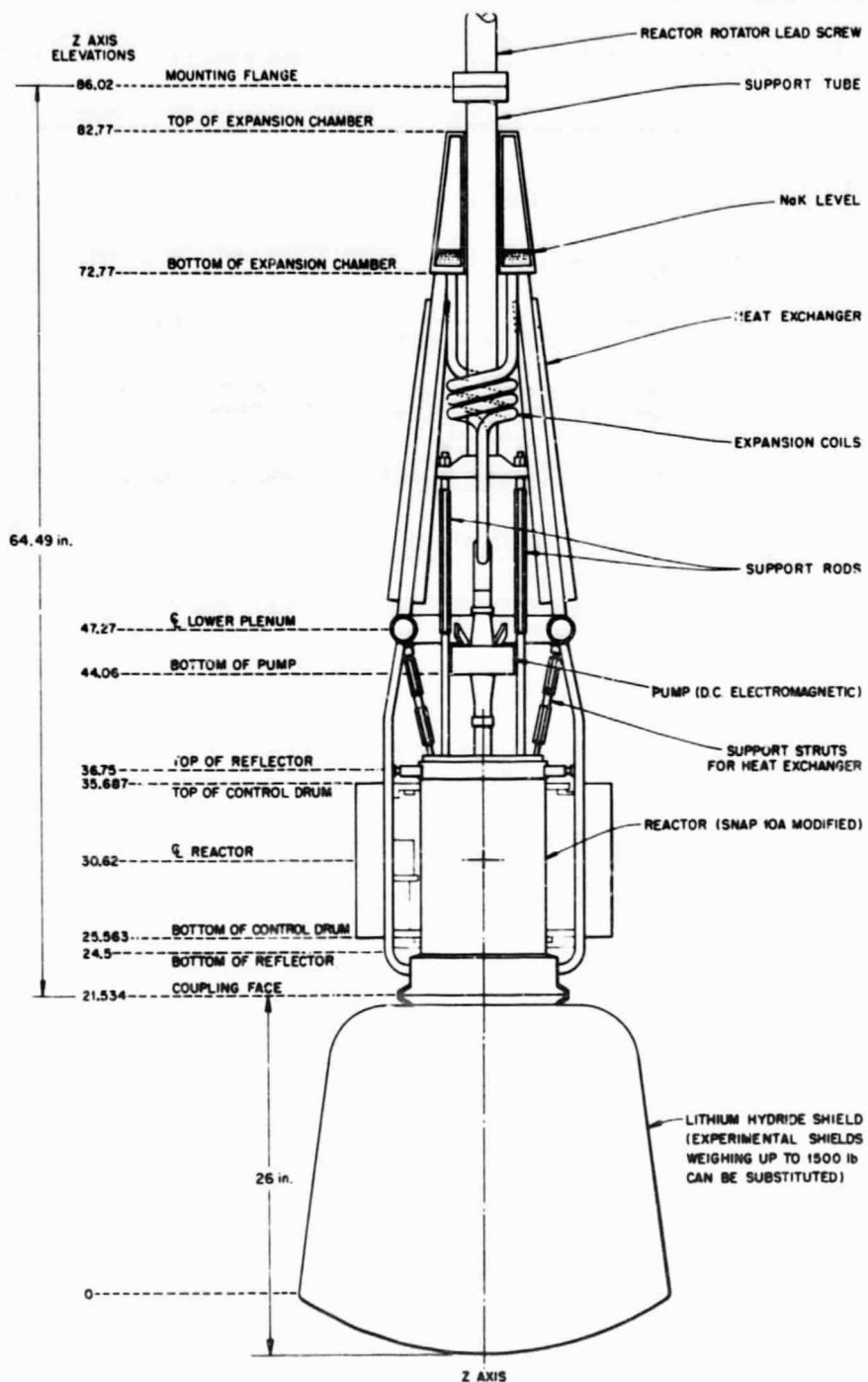
FIGURE 9 Drawing of the modified SNAP-10A reactor and lithium hydride shield used in TSF experiments.

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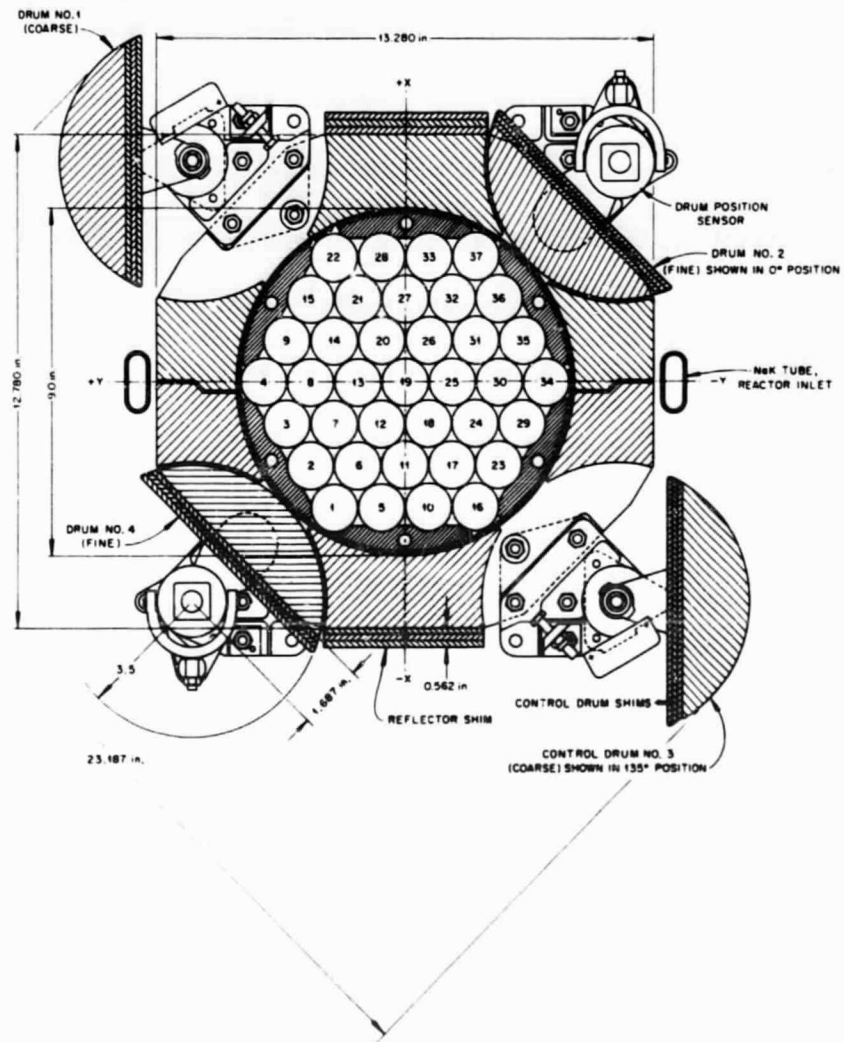
FIGURE 10   Plan view of modified SNAP 10-A reactor core showing rotating control drums.

CONCEPTUAL DESIGNS FOR 100-MEGAWATT SPACE RADIATORS

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ABSTRACT

A description and comparison of heat rejection systems for multimegawatt space-based power supplies is given. Current concepts are described, and through a common performance parameter these are compared with three advanced radiator concepts. The comparison is based on a power system that rejects 100 MW of heat while generating 10 MW of electrical power.

INTRODUCTION

Recently, increasing attention has been drawn to the importance of compact space power supplies in the successful pursuit of the space resource (Angelo and Buden, 1982). This resource has been identified as the deployment of military systems, extraterrestrial materials processing, creation of permanent space habitats, and planetary exploration. Attainment of these goals will require power sources in the multimegawatt range and larger. With power supplies in this size range, the heat rejection subsystem becomes the largest and heaviest component of the system. For this reason, new designs for large space heat rejection systems are needed.

As a basis for comparing various heat rejection systems, a baseline performance requirement was defined. In support of a 10-MW(e) power system, a 100-MW(t) heat rejection subsystem is assumed to be required. Consistent with this conversion efficiency is a heat rejection temperature of 1000 K. In addition to these performance requirements, the heat rejection subsystem must be capable of being transported into space either as a deployable unit or as modularized subunits that are assembled in space.

Several advanced radiator design concepts are presented, beginning with the most technically feasible system, a heat pipe radiator. Other concepts are presented that are more compact and lighter but are based on less well developed technologies. Finally, a performance parameter is defined that provides a means of comparing the various

systems discussed. The concepts are divided into two categories, those that have been proposed previously and new radiator designs that are in the very early stages of definition.

PRESENT CONCEPT

Conventional space radiator designs (MacKay   ) employ fin-tube geometries (Figure 1a). The coolant, usually a liquid, is pumped in a circuitous path through tubes that are connected to high thermal conductivity panels that act as fins or extended heat transfer surfaces. Fin-tube radiators are heavy because of the need for meteoroid protection around all of the tubing, but they have proven to be reliable. A significant improvement in performance can be achieved by utilizing heat pipes as isothermal fins (Figure 1b). However, the need for meteoroid protection remains, and the resulting weight savings is small.

Heat Pipe Radiators

Several radiator concepts using numerous, redundant heat pipes as fins have been proposed (Werner and Carlson, 1968, 1977) and built (Werner and Carlson, 1975). These systems utilize neat pipes connected to a central manifold containing a circulating coolant. The larger number of fin heat pipes eliminates the need for individual meteoroid protection on each heat pipe because the system is sized to accommodate a nominal loss of heat pipes due to meteoroid penetration. In addition, because of the nearly isothermal performance of the fins, their lengths can be increased, thus reducing the required manifold tubing that must have meteoroid protection. This configuration, shown schematically in Figure 2, results in a significant weight reduction compared with the conventional designs shown in Figure 1.

Both the conventional and the heat pipe radiators form a stationary structure that acts as the principal radiating surface for heat rejection. The next two systems to be discussed utilize a moving heat rejection surface.

Moving Belt Radiators

The moving belt radiator system consists of a moving heat transfer surface, a portion of which is in continuous contact with the reject heat source. Two example configurations (Weatherston and Smith, 1960, Schuerch and Robbins, 1964) are shown in Figures 3a and 3b. Both configurations require good thermal contact between the moving belt and the heated roller. The temperature drop across this interface at

high power levels could be significant, resulting in a severe
performance penalty.

Another concept (Hedgepeth, 1978) employing direct contact between
the working fluid and the moving belt is shown in Figure 4. In this
configuration, the contact resistance is eliminated; however, the
system requires a seal where the belt enters and exits the
liquid-metal heat exchanger. Development of a satisfactory seal for
the proposed operating conditions requires a substantial advance in
seal technology and design. Table 1 shows the parameters for a moving
belt radiator with a 100-MW(t) capacity. The moving belt radiator
combines a large radiating area with a deployable structure.

Particle Radiators

A second type of moving surface radiator is the solid-particle or
liquid-droplet radiator (Mattick and Hertzberg, 1980). The system is
shown schematically in Figure 5. Micron-size particles are heated in
a direct contact heat exchanger (Bruckner and Hertzberg, 1982). The
particles are then ejected to space from a "pitcher" and collected in
a "catcher." The duration of the trajectory is specified such that
the proper amount of heat is rejected from the particles to space.
The particles are then reheated, with subsequent release by the
"pitcher" in a continuous cycle. The density and size of the particle
stream is selected to optimize the radiant heat transfer. The
particle trajectory is determined by the direction and momentum
imparted by the "pitcher." This radiator system takes advantage of
the large surface area to volume ratio of small particles.

A variation of the dust particle radiator is the liquid-droplet
radiator, where liquid instead of solid particles are used. It has
further been suggested that latent heats of fusion be utilized in the
heat rejection system by letting the liquid droplets solidify in
transit. For the liquid system a droplet generator (Figure 6a) and a
liquid collector (Figure 6b) are required. The technique of
generating the droplets has been demonstrated with the development of
high-speed printers (Kuhn and Myers, 1979) in the computing industry.
Because of the zero-gravity environment, the collector must employ a
collection scheme, such as a spinning drum with a pump to recirculate
the radiator working fluid. The generator of artificial gravity for
the collector introduces additional complexity and increases the
radiator mass. In addition, creating a high-emissivity surface on the
liquid droplet is a problem requiring further development. Table 2
shows the physical parameters for a liquid-droplet radiator capable of
rejecting 100-MW. Because of the large droplet pathlengths required,
the structural weight of the liquid-droplet generator/collector system
is significant, which detracts from the weight savings inherent to
this concept. In the following section several additional proposed
high-potential heat rejection systems are described.

TABLE 1   Physical Parameters of Direct Contact Belt Radiator

Heat rejection temperature (MW)	100
Belt material	Graphite bonded to steel
Belt width (m)	11.6
Belt length (m)	131
Belt velocity (m/s)	15.2
Drum diameter (m)	0.9
Total mass (belt and drum) (kg)	7,260
Belt temperature (K)	560

TABLE 2   Physical Characteristics of Liquid-Droplet Radiator

Material	Tin droplet
Droplet diameter (μm)	50
Droplet emissivity	0.2
Droplet transit time (s)	3.45
Droplet temperature (K)	400
Sheet length (m)	58
Sheet height (m)	24
Droplet sheet thickness (cm)	4.8
Collector diameter (m)	1
Droplet spacing (μm)	250
M/Q (kg/kW)	0.017

PROPOSED CONCEPTS

Electrostatic Thermal Energy Radiator

The electrostatic thermal radiator (ETHER) is a new concept (data supplied by R. Hoeberling, Group AT-5, Los Alamos National Laboratory), which offers the promise of lower weight than the pitcher/catcher liquid-droplet radiator. The concept is shown schematically in Figure 7. The working medium is a liquid metal such as tin, which is formed into small (50-μm-diameter) droplets and subsequently charged to 10^{-13}-10^{-15}C (coulombs). The droplets, which are formed into a closely spaced (250-μm spacing) train, are ejected from the power system into space. Owing to an applied charge on the spacecraft that is opposite the droplet charge, the droplets

will execute a slightly elliptical orbit. If the ejection velocity is less than the escape velocity for the applied spacecraft charge, the particle stream will form a closed orbit. Thus the collector for the particles can be located on the spacecraft proper. This arrangement offers the promise of being lighter than an uncharged system because the boom and fluid channels necessary to support a mote collector are eliminated. During the orbit, each droplet radiates heat to outer space. In this heat rejection system, the time in orbit can be controlled by controlling the electrostatic potential between the droplets and the spacecraft. Another significant feature of this type of radiator is the possibility of charging the payload section of the spacecraft the same as the droplets to avoid contamination. A possible configuration for the system is shown in Figure 8.

Table 3 lists a typical set of parameters for the ETHER radiator for 100 MW of heat rejection.

Rotating Balloon Radiators

The rotating balloon radiator concept (data supplied by D. R. Koenig, Group O-13, Los Alamos National Laboratory) is one wherein a large radiator surface area is achieved by employing thin, lightweight materials such as graphite fiber bonded to steel in rotating radiator structures. One such system is shown schematically in Figure 9, and another embodiment is shown in Figure 10. In these concepts, liquid or vapor is ducted to the heat rejection surface by a central duct that also serves as support. When the heat transfer fluid is liquid, it is sprayed onto the rotating surface; when it is vapor, it condenses on the surface. Radiation to space cools the liquid as it is forced by the rotational body forces to flow to the outer extremities of the radiator, where liquid pumps are located. This concept offers the possibility of a light and flexible envelope structure that could be folded for launch. A second advantage is the ability to control the emissivity of the outer surface through the application of high-emissivity coatings. A disadvantage is the susceptibility to meteorite damage; however, it is not clear how much damage would be required to render this system inoperable.

For comparison to the previous concepts, Table 4 gives a set of calculated performance parameters for the rotating sphere radiator for 100-MW(t) heat rejection. An estimate of the mass of a rotating sphere radiator is presented in Table 5.

Filament Space Radiator Concept

The filament radiator concept discussed here (data supplied by M. A. Merrigan, Group Q-13, Los Alamos National Laboratory) is one wherein a continuous filament of a viscous material such as glass is drawn from a hot pool and caused to execute a trajectory in space prior to

TABLE 3 Parameters for a 100-MW(t) Electrostatic Thermal Radiator

Droplet material	Tin
Initial temperature (K)	1000
Droplet diameter (μm)	50
Droplet spacing (μm)	250
Droplet round-trip time (s)	4
Power station potential (kV)	450
Droplet mass (kg)	4.3×10^{-10}
Droplet initial velocity (m/s)	10

TABLE 4 Rotating Sphere Typical Parameters

Thermal power (MW)	100
Average rejection temperature (K)	1000
Surface emissivity	0.8
Working fluid	Sodium vapor
Envelope materials	0.25-mm carbon cloth bonded to 0.55-mm stainless steel foil
Total radiating area (m^2)	2,200
Sphere diameter (m)	26
Liquid flow rate (kg/s)	24
Average film thickness (laminar flow) (mm)	0.5
Fluid velocity at periphery (m/s)	0.41
Rotation frequency (rpm)	10
Centrifugal acceleration on periphery (g)	3.6
Sodium vapor pressure (Pa)	0.19×10^5 (2.8 psi)
Hoop stress (Pa)	$5,000 \times 10^5$ (73,000 psi)
Delta P pumps (Pa)	1.1×10^5 (16 psi)
Pumping power (kW)	3.4

returning to the spacecraft. The concept is one of a continuously renewable belt composed of a large number of very small (10-μm-diameter) filaments. A schematic of such a heat rejection system is shown in Figure 11.

The generation of small filaments gives a large surface area to volume ratio for this concept. As in the charged droplet concept, the

TABLE 5 Rotating Balloon Radiator Mass Summary (kg unless otherwise noted)

	Condensing Vapor	Liquid
Envelope	2,200	2,200
Liquid film	770	1,690
Liquid on periphery	230	1,150
Return pipe with liquid	180	1,620
Center shaft	400	810
Pumps	50	500
Drive motor	50	50
Structure (10%)	390	800
Total	4,270	8,820
Specific mass (kg/kW)	0.04	0.09

material exit and return points can be colocated on the spacecraft instead of at the opposite ends of a long boom. The concept offers the promise of low mass loss, although some problems with filament breakage need resolution. The physics governing the heat rejection for the filament radiator are very similar to those governing the liquid-droplet systems. Table 6 gives a list of typical performance parameters for 100 MW of heat rejection with a filament radiator. An estimate of the mass of a filament radiator system is shown in Table 7.

SPACE RADIATOR PERFORMANCE

To make a meaningful comparison of the various proposed systems, a performance parameter based on radiator mass per unit of heat rejection is proposed. Designating the parameter as z, we can write

$$z = M/Q$$

Heat rejection from the radiator surface is proportional to the fourth power of the radiating temperature,

$$Q = 2A\sigma\epsilon T^4$$

where both sides of the surface are considered active. Then the performance parameter is

$$z = M/2A\sigma\epsilon T^4$$

TABLE 6   Performance Parameters for a 100-MW(t) Filament Radiator

Total thermal capacity (MW)	100
Filament exit temperature (K)	1000
Filament return temperature (K)	300
Filament diameter (m)	7.5
Filament material (glass) flow rate (kg/s)	140
Filament velocity (m/s)	100
Transit time (s)	1
Total number of filaments	1.4×10^7
Number of nozzle banks required	14,000

TABLE 7   Filament Radiator Concept Estimated
Masses for A 100-MW(t) System (kg)

Filaments	140
Melt	500
Heat exchanger	500
Bushings	300
Collector	50
Total	1,490

$$Z = (1,490/1 \times 10^5) = 1.49 \times 10^{-2} \text{ kg/kW.}$$

If the surface emissivity can be made close to unity, then the above equation simplifies to

$$Z = (m/2A\sigma)(1/T^4$$

The first term in the above equation is designated the temperature coefficient and given the symbol C, so that

$$Z = C/T^4$$

Table 8 lists typical values of C for fin-tube, heat pipe, and particle radiators. The results from Table 8 are plotted in Figure 12 along with data for proposed radiation systems. Also, data for the

TABLE 8 Relative Performance of Radiator Types

Type	$C \left(\dfrac{kg - K^4}{kW} \right)$	$M/A \left(\dfrac{kg}{m^2} \right)$
Fin-tube	8.11×10^{11}	92
Heat pipe	1.76×10^{11}	20
Particle	6.97×10^{10}	7.9

Where $C = 8.82 \times 10^9$ M/A for the units indicated.

large-power advanced systems discussed in the section on proposed concepts are shown. At operating temperatures near 1000 K the performance variation between systems is a factor of about 150 based on Z. With such potential improvements in performance, additional studies are needed to determine the engineering feasibility of the proposed systems.

CONCLUSIONS

For space power systems operating in the multimegawatt regime, the heat rejection system becomes weight controlling. For example, a 100-MW(t) heat rejection system operating at 1000 K would have a mass of about 20,000 kg using heat pipe technology. The radiator mass for this example was taken from Figure 12. The entire system for a 10-MW(e) power level is estimated to be around 30,000 kg. If one of the advanced technologies could be developed, the radiator mass could be reduced to around 2,000 kg. The total weight for a 10-MW(e) system with an advanced radiator would be around 12,000 kg. Potential weight savings of this magnitude are a strong incentive to development of advanced space radiators.

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NOTATIONS

A   surface area of radiating panel.

C   temperature coefficient.

M   mass.

Q   heat rejection.

T   temperature.

    surface emissivity.

    Stefan-Boltzmann constant.
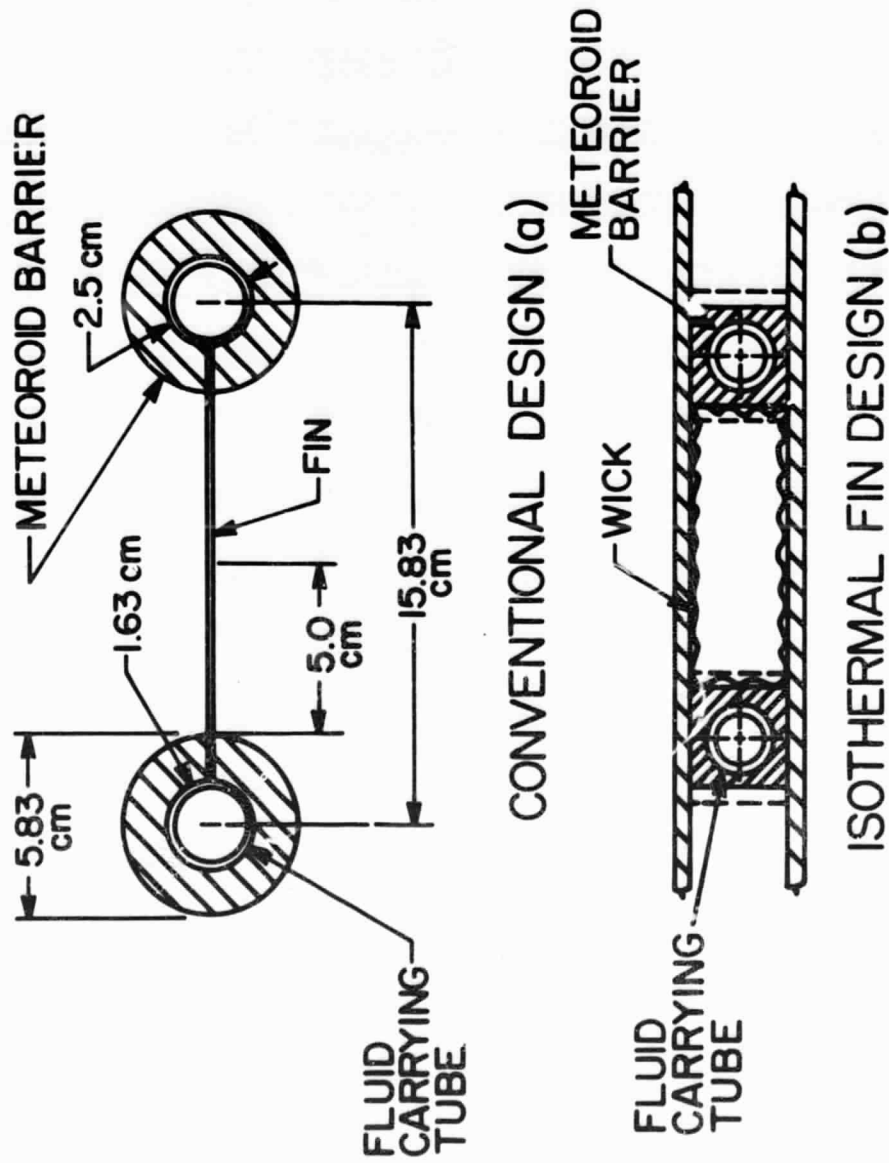
FIGURE 1 Fin-tube radiators.
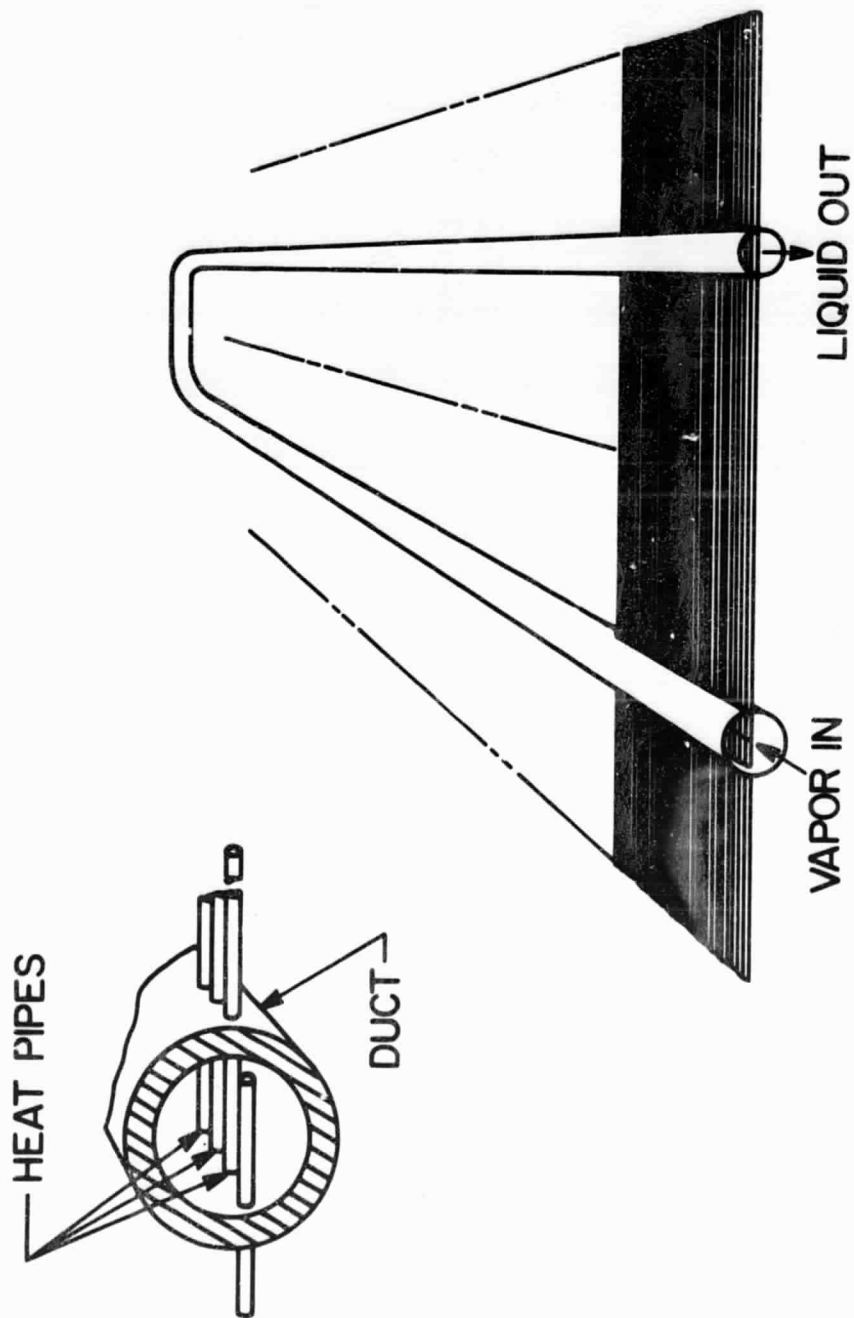
FIGURE 2 LLL heat pipe radiator.

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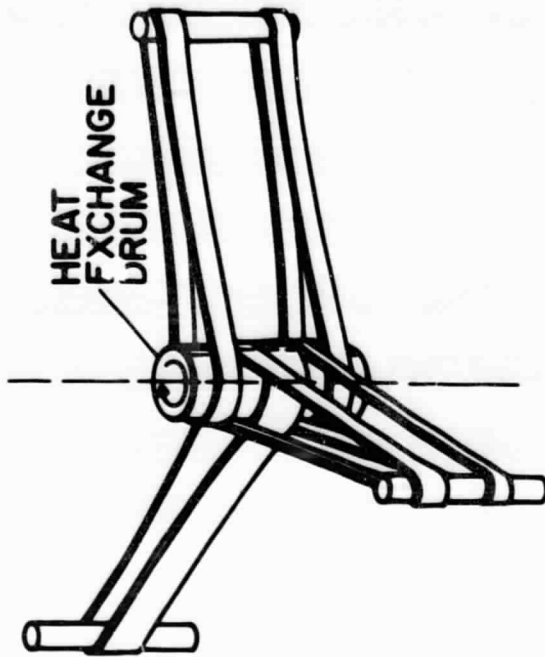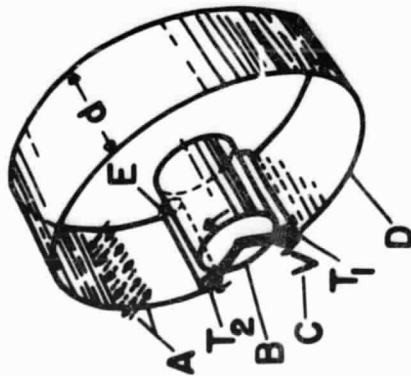
HEAT EXCHANGE DRUM

CONCEPT (b)

A. BELT LOSES HEAT BY RADIATION FROM BOTH SIDES
B. HEAT EXCHANGE DRUM
C. BELT SPEED
D. HEAT EXCHANGE BELT
E. GUIDE ROLLER

CONCEPT (a)

FIGURE 3   Direct contact belt radiator.
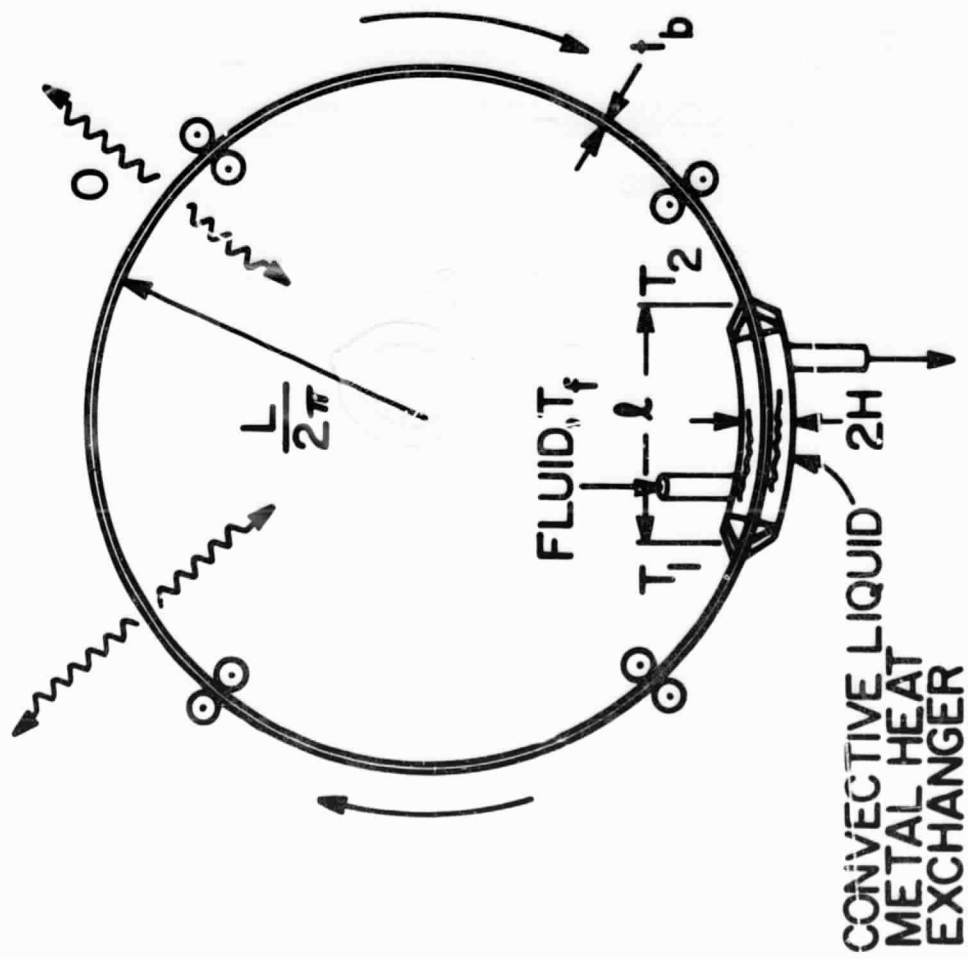
FIGURE 4   Convection-heated belt radiator.
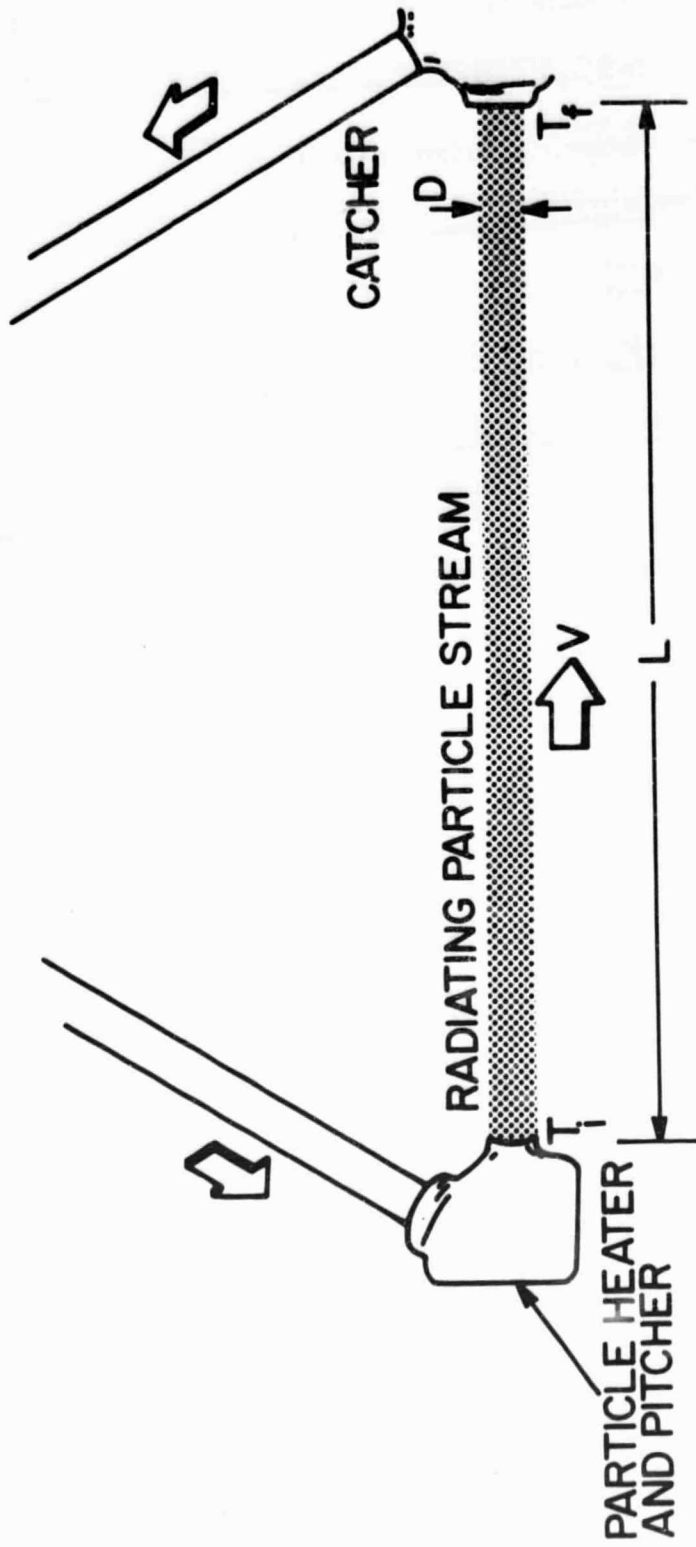
FIGURE 5   Dust particle radiators.

VIBRATOR

NOZZLE

DROPLET GENERATOR

(a)

PUMP

DROPLET STREAM

TO HEAT EXCHANGE

MAIN PUMP

ROTATING SEAL
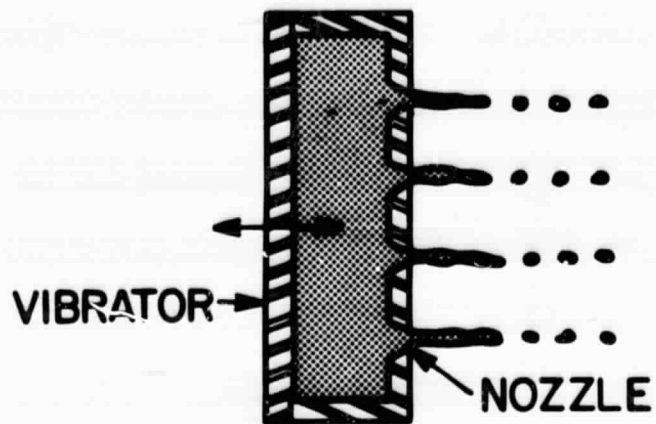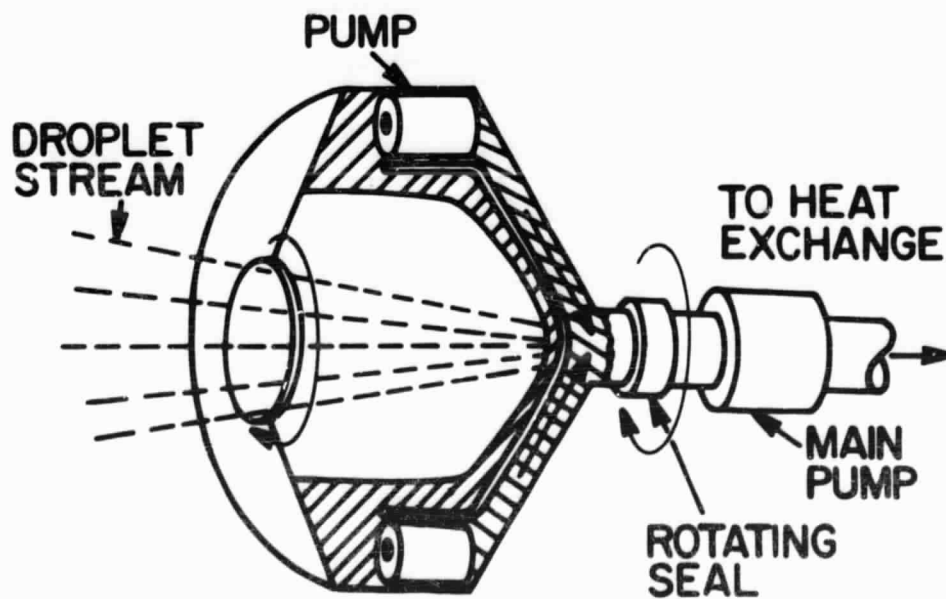
DROPLET COLLECTOR

(b)

FIGURE 6 Liquid-droplet-radiator components.
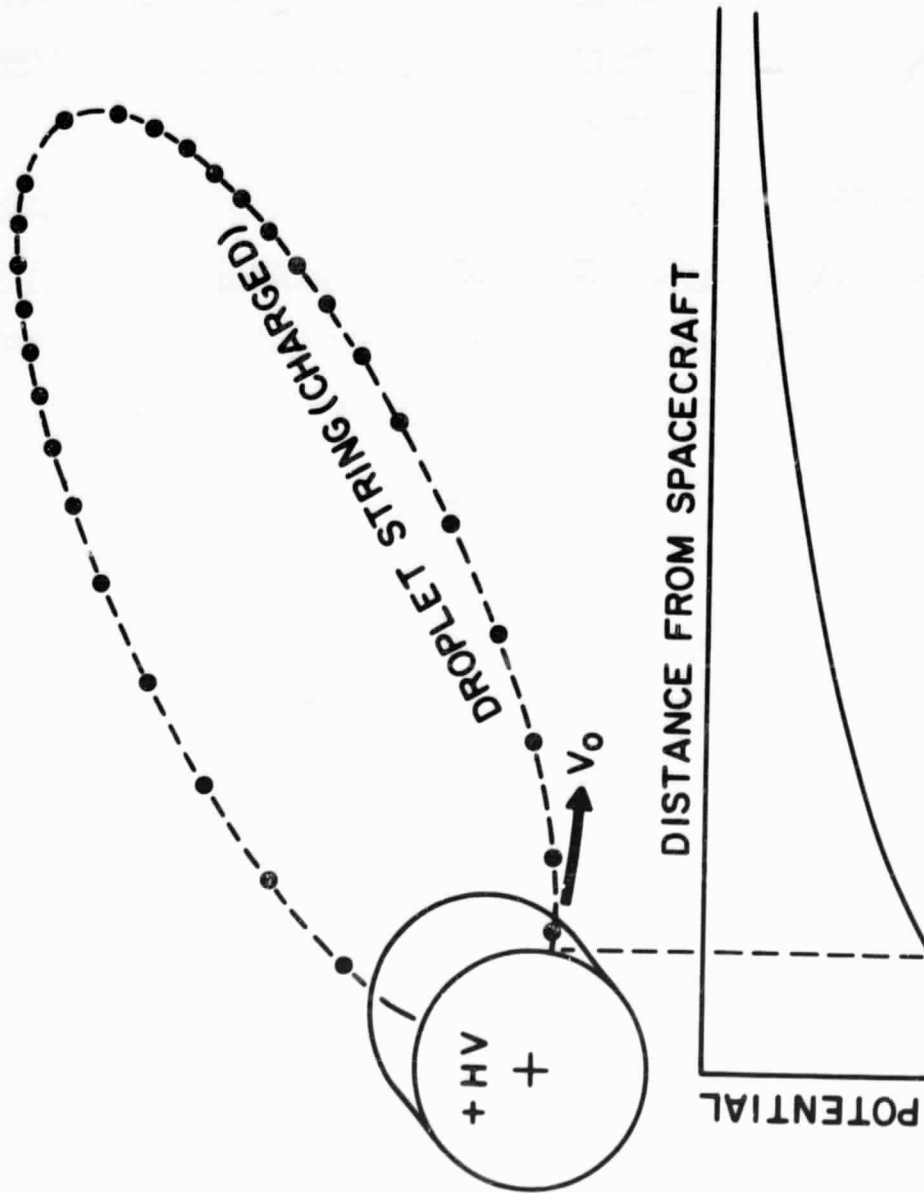
FIGURE 7 Schematic of the ETHER/concept.
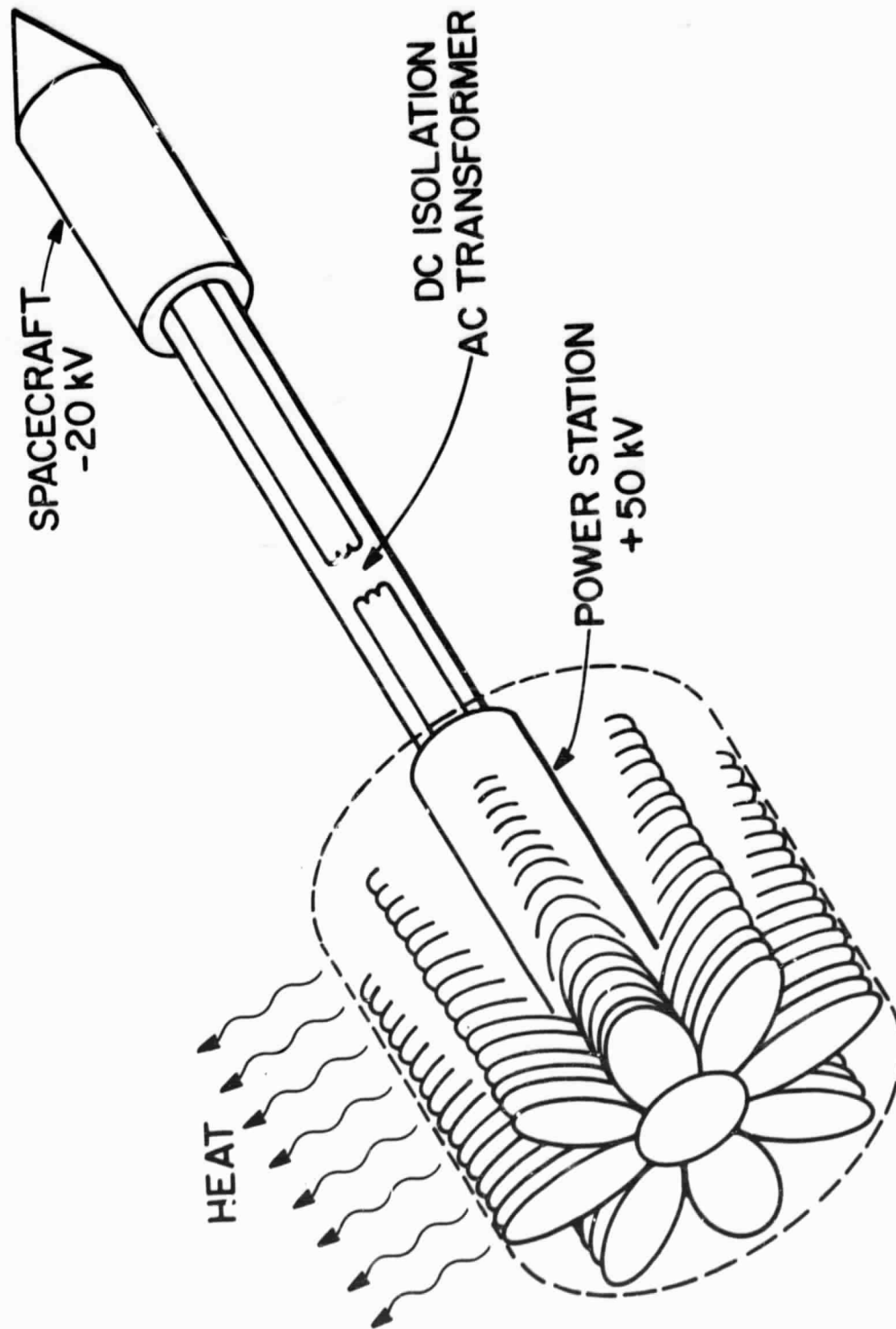
FIGURE 8   Possible ETHER configuration.
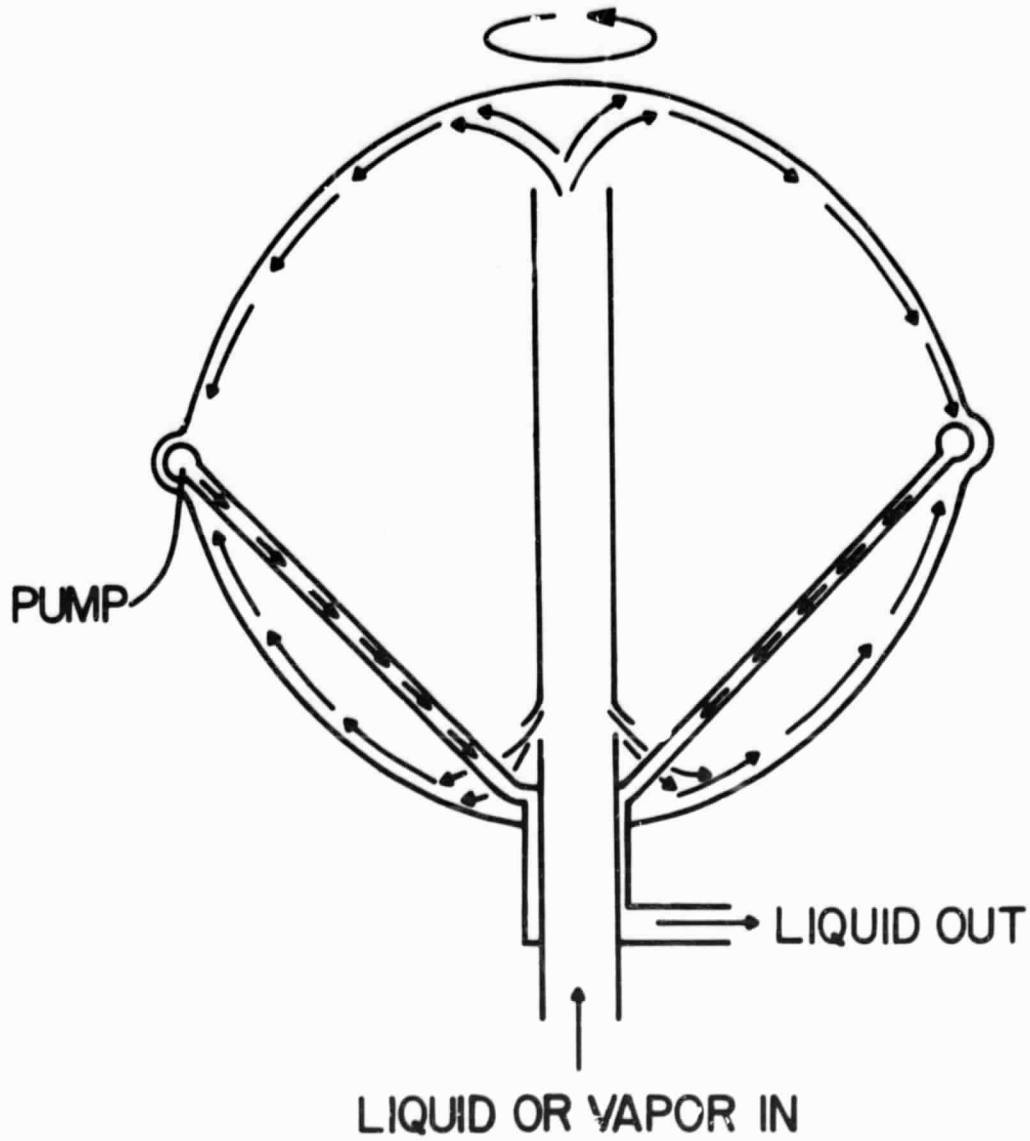
FIGURE 9 Rotating balloon (sphere) radiator.
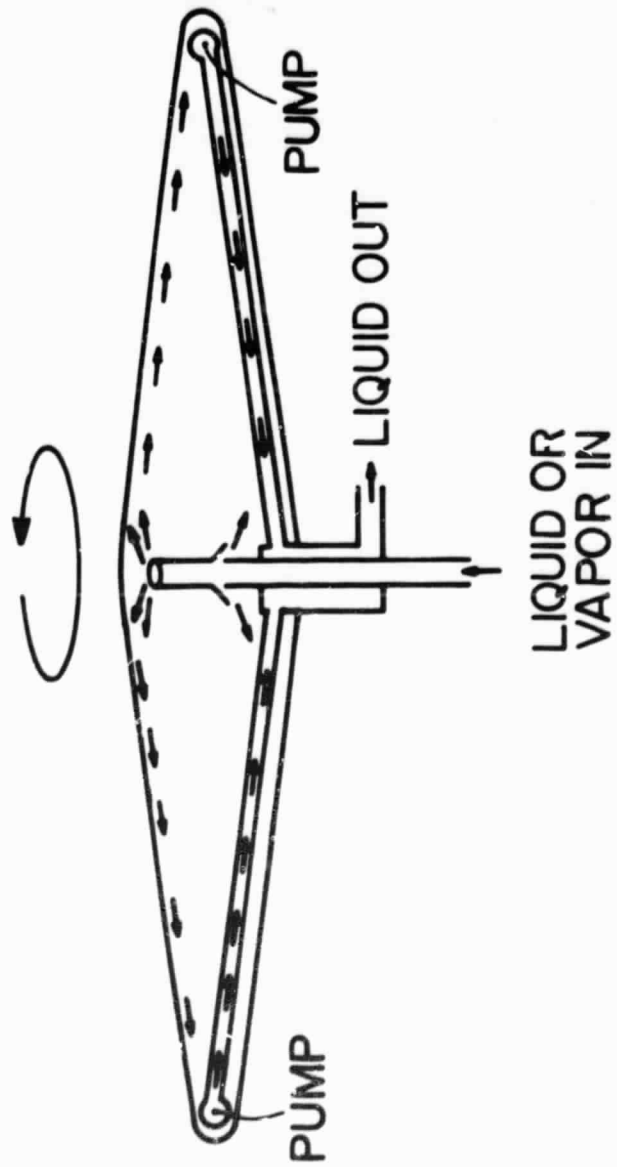
FIGURE 10 Rotating balloon (disk) radiator.
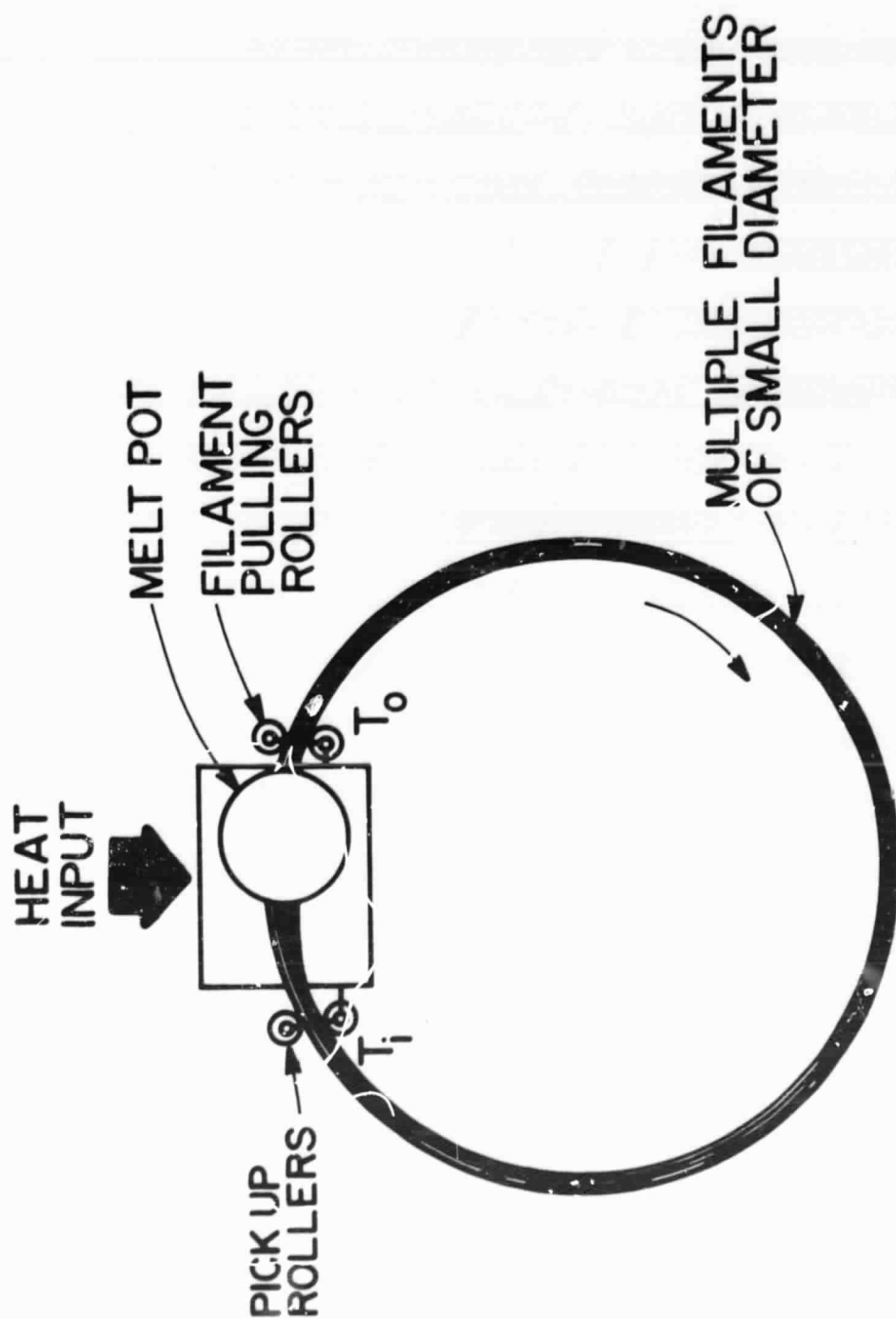
FIGURE 11 Schematic of a filament space radiator.
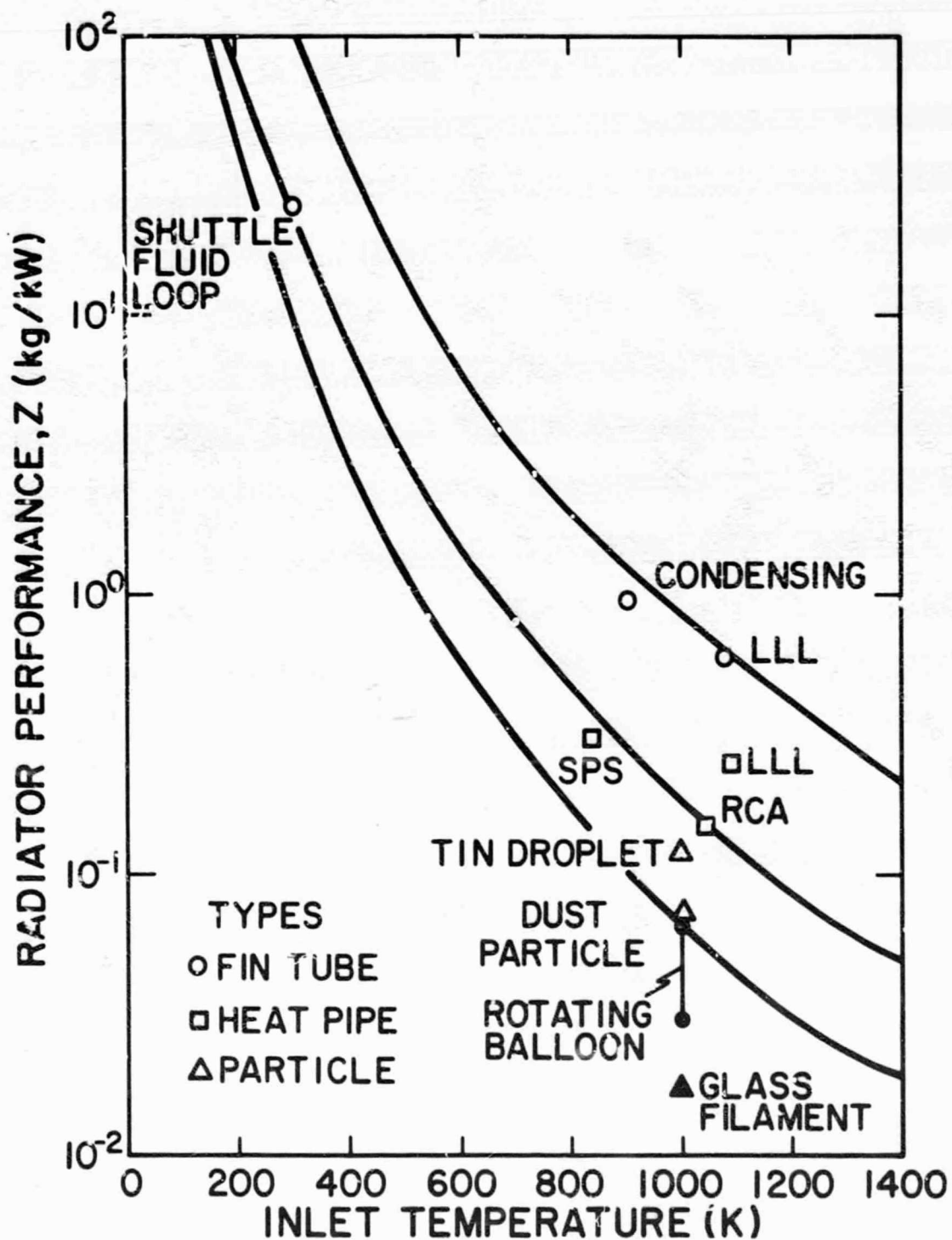
FIGURE 12 Comparison of advanced space radiator performance.

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POTENTIAL MISSION REQUIREMENTS
FOR COMPACT NUCLEAR REACTORS
IN TERRESTRIAL AND AERONAUTICAL APPLICATIONS

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ARMY REQUIREMENTS AS THEY RELATE TO
POSSIBLE USE OF COMPACT REACTOR SYSTEMS

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I have been asked to comment on the potential mission requirements for compact reactors in the Army. My comments will be very brief because, so far as I have been able to determine, the Army has no potential mission requirements for compact nuclear reactors.

However, the Army does have some encouragement for others who may be considering the development of compact nuclear reactors for military-related missions.

ARMY REACTOR PROGRAM

First, let me review the history of army development and use of compact nuclear reactors. The Army Nuclear Power Program (Table 1) was a Department of Defense/Atomic Energy Commission joint program (with the Army acting as lead agency) for developing nuclear reactors to serve military power needs on land. Of the operational plants shown in Table 1, PM-1 was for the Air Force (a remote radar site in Wyoming), PM-3A was for the Navy (in Anarctica), and the other four operational units were for the Army.

Notice that four of the reactors provided both electricity and steam for space heating. It has been suggested that the Army might, even today, have a potential requirement for developing nuclear reactors that cogenerate. The Army has already successfully done that.

The PM-2A Reactor

Two of the reactors are of special interest. The PM-2A was the first truly portable nuclear power plant in the world. Based on the proven design of the SM-1 at Fort Belvoir, the PM-2A provided electricity and steam for Camp "Century--The City Under the Ice"--in Greenland. Camp Century, built by the U.S. Army Corps of Engineers, was constructed entirely below the ice cap surface. The modular construction of the reactor was significant. The air-blast coolers, heat exchangers, electrical switchgear, turbine generator, condenser, and control

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TABLE 1 Summary of Army Nuclear Power Program

Reactor Designation	Location	Application	Date Critical	Date Operational	Date Decommissioned	Output, MW(e)	Reactor Type
Prototypes:							
SL-1	NRTS, Idaho	Testing, training	August 1958	October 1958	1962	0.2	BWR
SM-1	Ft. Belvoir, Virginia	Testing, training	April 1957	April 1957	1974	1.85	PWR
ML-1	NRTS, Idaho	Mobile power	March 1961	September 1962	1965	0.50	GCR(N_2)
Operational designs:							
SM-1A	Ft. Greely, Alaska	Base power and heat	March 1962	June 1962	1972	1.64	PWR
PM-1	Sundance, Wyoming	Base power and heat	February 1962	June 1962	1968	1.0	PWR
PM-2A	Camp Century, Greenland	Base power and heat	October 1960	February 1961	1963	1.5	PWR
PM-3A	McMurdo Sound, Antarctica	Base power and heat	March 1962	June 1962	1974	1.5	PWR
MH-1A	Gatun Lake, Panama Canal Zone	Barge-mounted mobile power	January 1967	January 1968	1976	10.0	PWR

SOURCE: W. R. Corliss, 1968, Power Reactors in Small Packages. Washington, D.C.: U.S. Atomic Energy Commission Division of Technical Information.

center were mounted on separate skids. The plant was designed to be air-transportable and assembled in the field within 90 days of arrival. (It was assembled, in fact, in 77 days.) Furthermore, it was designed to be removed and reassembled elsewhere. It was actually removed in 1964, back to the United States, where most components were placed in storage or used as spares; the nuclear core was used in the SM-1 reactor at Fort Belvoir.

The PM-2A is of special interest because it has been suggested that the Army might, even today, have a potential requirement for developing a modular nuclear reactor that could operate underground at some remote command post. The Army has already successfully done that. (Incidently, the PM-1 and PM-3A were also modular plants based on the SM-1).

The MH-1A Reactor

Another Army reactor of special interest was the MH-1A, a 10-MW floating nuclear power plant (Figure 1). Mounted in a liberty ship whose power plant had been removed, the MH-1A was towed to Gatun Lake in the Panama Canal Zone, where it provided electricity for several years under emergency conditions. It has been suggested that the Army might, even today, have a potential requirement for developing a barge-mounted nuclear power plant that could be towed to those areas of the world where an emergency need for electric power might occur. The Army has already successfully done that.

TECHNICAL FEASIBILITY

The Army has demonstrated the technical feasibility of developing small nuclear power plants to serve our defense needs for power on land. Nevertheless, the economic disadvantages of actually operating these nuclear plants in comparison with more conventional means resulted in the eventual decommissioning of all nuclear plants. That is, the job could be done just as well, and more cheaply, by non-nuclear means.

There may be requirements that the Army now has, or soon will have, in which the job cannot be done just as well by some non-nuclear means.

SPACE ASSETS

The Army has already begun to depend upon U.S. satellites for present operations and will become even more dependent in future years for the prosecution of "Airland Battle 2000," the current Army doctrinal concept of how the Army will fight (if called upon) near the end of this century.

Those satellites, which are not Army assets and for which the Army has no responsibility, need to be both more powerful and survivable. It may be that small nuclear power plants--and only nuclear power plants--can provide that survivable power.

Communications

The Army, through the U.S. Army Satellite Communications Agency, has already developed and fielded a family of ground communications systems that communicate via DSCS II. With DSCS II and MILSTAR, it is expected that satellites will operate with greater beam directivity and with higher power densities and with improved antijamming capability. That will enable our Army receivers and transmitters to be small, have smaller antennas, and have more channels available.

Weather Forecasting

The Defense Meteorological Satellite program is another operational system that will be improved so as to provide tactical commanders timely annotated weather maps or reports for use in connection with the threat (or actual use) of smoke, chemical agents, or nuclear weapons by the enemy.

Positioning

Land navigation and accurate positioning of both friendly forces and the enemy have long been imperfectly attained. The Global Positioning System (or Navstar) may meet many of our needs.

War-Zone Monitoring

War-zone monitoring by satellites equipped with sensors to detect chemical and nuclear weapons use are a possibility. Ground terminals located in the war zone could receive data from the satellites indicating the contaminated areas on the battlefield. The launch of tactical and medium-range missiles from the enemy second echelon could be detected, and the missiles might be tracked.

Readout of Remote Senors

Satellite readout of remotely implanted battlefield sensors could allow location of second echelon and reserve forces, or of chemically or radiologically contaminated areas.

Mapping

Landsat-type imagery could be used for engineering support to quickly assess areas for soil trafficability and for selection of routes through swamps, flooded areas, forests, etc.

Survivability

As the Army comes to depend upon these space assets more and more, it becomes more and more necessary that those assets be there when they are needed and that they can be seen and heard by us. They need to be more powerful and jam-proof, and above all they need to be capable of surviving enemy antisatellite efforts. Most U.S. satellites depend on solar cells for power. These cells may not provide enough power for a jam-proof satellite and they may not be made survivable enough.

The United States had a space power reactor program for many years, but of the 22 nuclear power space applications between 1960 and 1980, only one was a nuclear reactor; the rest were radioisotope thermoelectric generators. Table 2 shows that relatively high power nuclear reactors for space applications have been developed, but that only SNAP-10A has been flown.

SUMMARY

In summary, the Army has successfully demonstrated that small nuclear reactors can be developed and used in military applications on land. However, at the time each was decommissioned, continued operation could not be justified economically.

The Army is, however, developing a dependence upon U.S. space assets which the Army does not own. The Army needs those satellites to be, in many cases, more powerful and, in all cases, survivable. It may be that small nuclear power plants should be developed to provide that power.

TABLE 2   U.S. Space Power Reactor Programs to Date

	SNAP-10A	SNAP-2	SNAP-8	SNAP-50	Thermionic	SPAR
Time period	1962-1965	1957-1964	1960-1970	1962-1965	1965-1972	1977-1981
Power level (kW(e))	0.5	3-5	35	36/300	50 & up	100
Reactor type	Thermal	Thermal	Thermal	Thermal	Both	Fast
Fuel	UZrH	UZrH	UZrH	UN	UO_2 UC/ZrC	UO_2
Fuel shape	Cylinder	Cylinder	Cylinder	Cylinder	Cylinder	Wafer
Fuel clad	Cylinder	Cylinder	Cylinder	Cylinder	Cylinder	None
Coolant pipes	Stainless steel	Stainless steel	Stainless steel	Nb-1Zr	Nb-1Zr	Mo or Mo/Re
Coolant	NaK	NaK	NaK	Li	Na,NaK	Na or Li
Heat transport to converter	Pumped NaK	Pumped NaK	Pumped NaK	Pumped Li	Conduction	Heat pipes
Conversion system	Thermo-electric	Rankine (Hg)	Rankine (Hg)	Rankine (K)	Therm-ionic (Cs)	Thermo-electric (SiGe/GaP)
Reactor output temperature K	830	920	980	1370	1650-2100	1450
Heat transfer to radiator	Conductive	Vapor	Pumped	Pumped	Pumped	Heat pipes
Component feasibility demonstrated	Yes	Yes	Yes	No	No	No
Reactor operated on ground	10^4 h	10^3 h	1 year	No	No	No
Reactor operated in space	10^3 h (S/C failure)	No	No	No	No	No
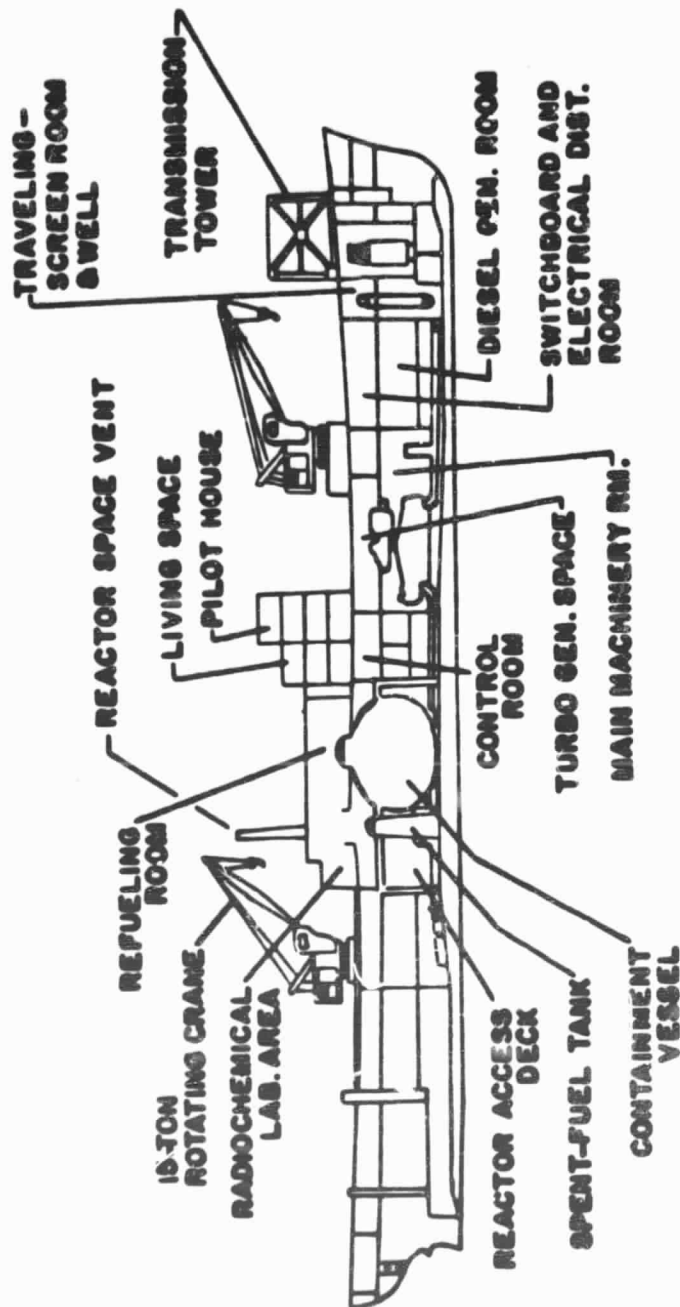
FIGURE 1 The MH-1A reactor as installed aboard the liberty ship STURGIS.

ADVANCED NAVAL AIR VEHICLES:
NUCLEAR CONSIDERATIONS

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- THIS BRIEFING WILL DEAL SOLELY WITH APPLICATION OF NEW PROPULSION TECHNOLOGY TO AIRCRAFT

- NEW PROPULSION SYSTEMS ARE NEEDED FOR ADVANCED NAVAL AIR VEHICLES

- LOW MASS NUCLEAR POWER SYSTEMS ARE SERIOUS CANDIDATES FOR AT LEAST ONE OF THE ADVANCED A/C CONCEPTS

FIGURE 1   Agenda.

- THERE IS A LIMITED TECHNICAL BASIS FOR AIRBORNE NUCLEAR POWER

- SAFETY IS THE PARAMOUNT ISSUE FOR BOTH THE PUBLIC AND THE MILITARY. THE SAFETY ISSUE HAS BECOME POLARIZED IN THE MINDS OF MANY PEOPLE. SAFETY ISSUES WILL DOMINATE:

  - DESIGN

  - TESTING

  - BASING

  - FLIGHT PATHS

  - COST OF DEVELOPMENT

- NAVY USE FROM COASTAL BASES WITH OVER-WATER FLIGHT PATHS WOULD AVOID MUCH OF THE EMOTIONAL ASPECTS OF THE SAFETY ISSUE

FIGURE 2 Premise.

- NUCLEAR PROPULSION FOR ADVANCED AIR VEHICLES APPEARS APPROPRIATE ONLY FOR VEHICLES ABOVE A CERTAIN MINIMUM SIZE – PROBABLY OVER 200 TONS

- THERE IS A LIMITED TECHNOLOGY BASE FOR THE DEVELOPMENT OF LIGHTWEIGHT (LOW MASS) NUCLEAR POWER THAT COULD BE EXPANDED

FIGURE 3   Current status.

● NUCLEAR PROPULSION FOR ADVANCED AIR VEHICLES APPEARS

  APPROPRIATE ONLY FOR VEHICLES ABOVE A CERTAIN MINIMUM

  SIZE – PROBABLY OVER 200 TONS

● THERE IS A LIMITED TECHNOLOGY BASE FOR THE DEVELOPMENT

  OF LIGHTWEIGHT (LOW MASS) NUCLEAR POWER THAT COULD

  BE EXPANDED

FIGURE 4   Current status (continued).

TO ENHANCE THE ROLES OF ADVANCED NAVAL AIR VEHICLES

- **INCREASE RANGE**

- **EXTEND ENDURANCE**

- **REDUCE DEPENDENCE ON PETROLEUM FUELS**

- **REDUCE DEPENDENCE ON OVERSEAS BASES**

- **IMPROVE PAYLOAD CAPABILITIES**

- **IMPROVE MOBILITY AND FLEXIBILITY OF OPERATIONS**

- **REDUCE COST OF ACQUIRING INCREASED MILITARY CAPABILITIES**

FIGURE 5 Key objectives of advanced naval air vehicle systems design.

SIZE OF CONVENTIONAL, FOSSIL-FUELED, TURBOPROP AIRCRAFT INCREASES RAPIDLY AS ENDURANCE EXCEEDS ONE DAY

LIQUID-HYDROGEN IMPROVES ENDURANCE BUT LOW DENSITY OF LH_2 LEADS TO FUSELAGES OF MUCH LARGE VOLUMES. NET RESULT IS AIRBORNE ENDURANCE OF 2 TO 3 DAYS. USE OF LH_2 WOULD REQUIRE A WORLD-WIDE DISTRIBUTION SYSTEM MUCH LIKE THAT FOR PETROLEUM FUELS BUT SEPARATE.

NUCLEAR POWER OFFERS POTENTIAL FOR INDEFINITE ENDURANCE. LIMIT IS SET BY CREW AND RELIABILITIES.

APPLICATION OF NUCLEAR POWER TO AIRCRAFT SHOULD CONSIDER THEIR FUTURE EMPLOYMENT, USING ADVANCED AVIATION CONCEPTS SUCH AS -

- AIR LOITER A/C ● SEA LOITER A/C ● LTA VEHICLE ● WIG VEHICLE

FIGURE 6   Design factors for advanced aircraft.

- NEED FOR IMPROVED FUEL EFFICIENCY IN BOTH AIRCRAFT AND MARINE ENGINES WELL RECOGNIZED IN 1970's

- PROJECTED (1976 STATE OF THE ART) FUEL EFFICIENCY IMPROVEMENTS OF 10 TO 40 PERCENT BY MID-1980's

- NEW GENERATION "BIG FAN" — HIGH-BYPASS-RATIO POWER PLANTS IN 40,000 LB THRUST RANGE — ARE 15 TO 25 PERCENT BETTER THAN PREDECESSORS

- PROSPECT FOR FURTHER SIGNIFICANT IMPROVEMENT IN SPECIFIC FUEL CONSUMPTION OF CONVENTIONAL ENGINES IS LIMITED

FIGURE 7 Propulsion technology.

- CURRENT FORWARD RESUPPLY CONCEPTS FOR CRITICAL ITEMS ARE DEPENDENT UPON OVERFLIGHT RIGHTS, AVAILABILITY OF LANDING RIGHTS AND REFUELING AT OVERSEAS AIR BASES

- MILITARY AIRLIFT COMMAND SUPPORT IS CURRENTLY TAILORED TO SUPPORT OF NATO AND THE FAR EAST

- IF GROUND FUEL IS NOT AVAILABLE, THE REQUIREMENT FOR INFLIGHT REFUELING DOES NOT SATISFY NAVY REQUIREMENTS AS SHOWN IN THE FOLLOWING VUGRAPHS

FIGURE 8   Operational problem.

- LOSS OF OVERSEAS BASES AND POLITICAL ALIGNMENTS

- AUGMENT SEA LINES OF COMMUNICATIONS IN DELIVERY OF RAW MATERIALS

- EXPANDING THREAT - CAPABILITY AND AREA

- DECREASE IN AVAILABILITY OF PETROLEUM

- INCREASING COSTS OF R&D, ACQUISITION, AND OPERATION

- EXTENDED AT-SEA PERIODS IN REMOTE AREAS WITHOUT NEAR BY SUPPORT FACILITIES

- RELATING LOSS OF ABILITY OF MILITARY TO INFLUENCE COMMERCIAL VEHICLE DEVELOPMENT

- POTENTIAL NON-AVAILABILITY OF PETROLEUM FUEL

FIGURE 9 Changing circumstances.

- U. S. CIVIL AND MILITARY INTERESTS IN FUTURE MAY DEPEND MORE ON THE AVAILABILITY OF FUEL AT AIRFIELDS AND SEAPORTS OUTSIDE CONUS THAN ON THE BASIC COST OF FUEL. (COST ESTIMATES ON FOLLOWING GRAPH)

- FUEL AVAILABILITY IS AN OLD ISSUE TO NAVY AND COMMERCIAL SHIPPING. IT WAS A KEY ISSUE WHEN SHIPS CONVERTED FROM

  - SAIL TO COAL
  - COAL TO OIL
  - OIL TO NUCLEAR POWER

- THE PRESIDENT AND CONGRESS RECOGNIZED THE SERIOUSNESS OF THIS ISSUE AND SUPPORTED THE NAVY WITH CONTIGENT FUEL CONTRACTS AT COAL AND OIL PORTS THROUGHOUT THE WORLD

FIGURE 10 Fuel availability.

OBJECTIVE: ....TO EXAMINE....TECHNOLOGY NOW BEING DEVELOPED.....EVALUATE

EACH POTENTIAL ADVANCED NAVAL AIR VEHICLE IN...LIGHT OF

TECHNOLOGICAL FEASIBILITY, AFFORDABILITY, AND APPLICABILITY TO

....EXISTING OR PROJECTED NAVAL MISSION.....INCLUDE EXAMINATION

OF ALL POTENTIAL NAVAL VEHICLES USING ADVANCED TECHNOLOGY

....CONSIDER....

(1) LONG RANGE, LONG AIR OR SEA LOITER

(2) WING-IN-GROUND AIRCRAFT

(3) LIGHTER-THAN-AIR VEHICLES

FIGURE 11   Advanced naval air vehicle concept evaluation (ANVCE).

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AIR LOITER VEHICLE -

● MORE THAN 3 DAYS ENDURANCE WOULD REQUIRE
THE USE OF LIGHTWEIGHT NUCLEAR PROPULSION
(LWNP)

● LWNP ASSUMED TO BE FEASIBLE AND USED IN
POINT DESIGN FOR NUCLEAR POWERED, AIR LOITER
A/C

FIGURE 12   Nuclear aspects of ANVCE recommendations.

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- THESE OBSERVATIONS CONFIRM WHAT IS GENERALLY ACKNOWLEDGED, THAT CONVENTIONAL POWER PLANTS AND PROPULSION SYSTEMS HAVE LIMITS SET BY THE LAWS OF PHYSICS

- THESE LIMITS IMPACT ON THE DESIGN OF ADVANCED AIR VEHICLES AND ON THEIR OPERATIONAL EMPLOYMENT. POINT DESIGNS REPRESENT COMPROMISES TO OPTIMIZE PARTICULAR CHARACTERISTICS SUCH AS RANGE OR SPEED

FIGURE 13   Comments.

- THE GUIDELINES TO OPTIMIZE ADVANCED AIR VEHICLES ARE THE OPERATIONAL REQUIREMENTS THAT ARE DRIVEN BY SUCH FACTORS AS THREAT, ESTIMATES OF PROBABLE AREAS OF OPERATION, PAYLOAD REQUIREMENTS (ORDNANCE OR LOGISTICS), AND PRESUMED AVAILABILITY OF BEANS, BULLETS, AND BLACK OIL

- AVAILABLE TECHNICAL EVIDENCE SUPPORTS CONCLUSION THAT IMPROVEMENTS IN ADVANCED AIR VEHICLES PERFORMANCE PARAMETERS ARE BECOMING MARGINAL IN MAGNITUDE, ADDITIONAL MILITARY CAPABILITY, AND COST SO LONG AS POWER PLANTS ARE LIMITED TO CONVENTIONAL FOSSIL FUELED TYPES

FIGURE 14 Comments (continued).

- THE FULL EXPLOITATION OF THE UNIQUE QUALITIES OF VARIOUS ADVANCED AIR VEHICLE DESIGNS APPEAR TO HAVE BEEN CURTAILED SO FAR BY THE PROPOSED USE OF PETROLEUM FUELED PROPULSION

- INSTEAD OF EXPLOITING THEIR UNIQUE CAPABILITIES TO DEVELOP NEW WAYS TO ACCOMPLISH NAVY MISSIONS, THEY HAVE GENERALLY BEEN LOOKED AT AS AN ALTERNATIVE TO CONVENTIONAL VEHICLES AND EMPLOYED IN THE SAME WAY

FIGURE 1 Comments (continued).

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- ANY FEASIBILITY STUDIES SHOULD ALSO CONSIDER

  - PUBLIC SAFETY - FLIGHT PATHS, BASING, HAZARD TO ENVIRONMENT

  - PERSONNEL SAFETY - RADIATION, CONTAINMENT, CASUALTY CONTROL IN FLIGHT

  - MAINTENANCE AND REFUELING - PERSONNEL, RADIATION EXPOSURE, NUCLEAR WASTE, FACILITIES, TRAINING

FIGURE 16 Comments (continued).

NUCLEAR POWER FOR MX MISSILE DEEP BASING

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On October 2, 1981, the President of the United States disapproved the horizontal, multiple protective shelter basing scheme for the MX missile and directed the Air Force to consider other basing options. The Air Force established three major studies to address basing MX in aircraft, in closely spaced silos ("Dense Pack"), and in tunnels deep underground. The purpose of this paper is to discuss the preliminary results of a power system study performed by the Air Force Weapons Laboratory (AFWL), Oak Ridge National Laboratory (ORNL), and Los Alamos National Laboratory (LANL). This study is addressing the crucial issue of how nuclear power might provide adequate, survivable power for the deep basing complex.

Some discussion of the rationale of the deep basing concept is necessary to understand the results of the power study. Soviet ICBM accuracy has progressed to the point that our Minuteman missiles are extremely vulnerable, and we must seek extraordinary basing options to guarantee survivability to first strikes. One method to achieve survivability is to place our missiles at a location that allows adequate "stand-off" from the effects of incoming weapons and thereby increases the survival probability of the system. Given Soviet CEPs (circular error probable), the only way to achieve the requisite stand-off is by placing the missiles deep underground with the proper covering geology to optimize attenuation of the weapon-induced ground shock. Deep basing as a concept also provides the ground-based leg of the strategic offensive triad an enduring secure reserve force that would be lacking in the Dense Pack concept. Deep basing would have up to a one-year survivability and could put at great risk Soviet population and industry. It would provide megatonnage exchange ratios with the Soviet Union at least 10 times that of the multiple protective shelter concept (Keltner and Diament; 1982).

An illustration of how a Deep Basing site would look is shown in Figure 1. Missiles, support facilities, and egress devices would be placed approximately 2,000 ft under a large mesa in hardened tunnels. The complex would operate in some ways like a land-locked submarine. That is, the facility must be able to "button up," support itself internally for up to one year, and finally egress missiles to launch.

button-up period to ensure equipment availability, maintain crew
training, and confuse the enemy as to actual egress timing (Peters,
1982). Our thinking about the deep basing concept centered around the
squadron as a basic unit. Each squadron would have men and equipment
as shown in Figure 10.

Power requirements are obviously very sensitive to assumptions at
this point. Figure 11 is our estimate of the power requirement per
squadron for this facility. The 0.45 is a power factor to account for
the fact that all power consumption devices would not be operational
simultaneously. The 4.5-MW requirement for missile launchers and
other TEL equipment is keyed to the inclined egress concept. This
requirement is a strong function of the angle of the egress tunnel.
The 5-MW total requirement is significant in that it means such power
concepts as fuel cells would have enormous reactant storage
requirements, which would be difficult to make safe and survivable to
the assumed weapons threat. This requirement is also significant in
that it is a small requirement for such concepts as nuclear reactors
and could probably be met with small, safe, and relatively simple
reactor concepts.

Before presenting the results of some of our studies, some
preliminary caveats are in order. First, this paper is intended to
illustrate how compact reactors may be useful to special DOD
applications, and not to advocate deep basing or nuclear power.
Second, the Air Force Weapons Laboratory (AFWL), Oak Ridge National
Laboratory (ORNL), and Los Alamos National Laboratory (LANL) were
tasked by the BMO to assess the feasibility of nuclear power for the
deep basing concept. They could not, within the modest scope of this
study, address the best system or develop point designs. Much work
remains to convert their ideas to real designs. Third, we from the
start assumed that it would be impossible or too expensive to design
and construct a single superhardened power facility capable of
withstanding the expected nuclear weapons environments. We therefore
approached the study with the assumption that system survivability
could also be obtained by a series of redundant, interconnected power
facilities with a survivable power distribution network.

An example of how the power distribution system could be configured
to achieve survivability is illustrated in Figure 12. This
"distributed hybrid" system, conceived by Lt. David Peters, AFWL,
could provide the required power system survivability (Peters, 1982).
The types of power plants are unspecified, but might be a combination
of nuclear reactors or reactors with fuel cell plants. Such
distribution systems would be expensive, since very sophisticated
electromagnetic pulse (EMP) protection would also have to be provided
against the large expected EMP environment even at 2,000 ft of depth.
Our study indicates that the distribution system may be the most
difficult part of the overall power survivability question.

Los Alamos National Laboratory was tasked to address light water
reactor concepts that would be feasible for the deep basing power
system. Their study (Los Alamos National Laboratory, in preparation)

displays much creative thinking about survivable light water reactors (LWR). LANL workers concentrated on the following four LWRs, on the assumption that an early start date for the deep basing system meant no new reactor concepts could be developed:

o   Naval reactor
o   Down-sized commercial reactor
o   Army Package Power Reactor (APPR)
o   Horizontal pressure tube reactor.

Naval reactors were of great interest to the study group, but since the Navy would release no survivability data to LANL, and since these reactors might not be available due to Navy priorities, naval reactors were not selected as the baseline design for the study. The Army Package Power Reactor was found to be simple, portable, relatively inexpensive ($333 million), and easy to build in modular plant concepts. Some of LANL's findings are included in Figure 13. Figures 14 and 15 illustrate one concept to fit the reactor in the prescribed 18-ft tunnel and how such modular reactors could be connected to form a redundant, survivable power system. From the results of this study, I conclude LWRs are a feasible, inexpensive (with regard to total life-cycle costs) option to power deep basing of missiles.

Oak Ridge National Laboratory was tasked to study the feasibility of using high-temperature gas-cooled reactors (HTGR) for powering this concept. Again, survivability issues led the study group independently to a small, modular, redundant reactor concept. Figure 16 (Oak Ridge National Laboratory, in preparation) summarizes the ORNL findings. Figures 17 and 18 are schematics of how the system might look. The ORNL workers chose a nominal power level of 15 MW(e) as reasonable, trading off power requirements, redundancy, and cost. A vertical in-line design was chosen so as to minimize piping connections (for survivability) and enhance natural circulation for cooling in case of loss of helium circulators. A steel pressure vessel was selected in lieu of a prestressed concrete vessel to allow the system to fit in the alloted 18-ft tunnel. The fuel selected was highly enriched ^{235}U encased in 6-cm-diameter graphite "pebbles" (Oak Ridge National Laboratory, in preparation). The reactor would consist of approximately 86,000 such "pebbles," randomly stacked. This concept is very similar to that embodied in the operational AVR reactor in the Federal Republic of Germanyy. This core design will enhance survivability in that displacements to the reactor vessel will not destroy the core geometry. ORNL has estimated overall cycle efficiency at 30 percent or greater for their base-line design. A key feature of this HTGR design is that it has such a large negative temperature coefficient that the operators can literally walk away and the reactor will be safely shut down. I consider this a key feature for a facility subject to nuclear weapons attack. The ORNL concept offers great promise and should be vigorously pursued if deep basing is selected as a concept.

The Air Force Weapons Laboratory is participating in this study by addressing the questions of reactor component survivability and the process by which the Air Force might gain approval to build and operate a nuclear reactor for this facility. We have found that little data exists for power components in these rather severe environments. From limited data associated with the Safeguard ABM System, we have tentatively concluded that without special shock isolation techniques, almost all power components would fail in these environments. AFWL has only scratched the surface in this area, and much work is left to be done.

With regard to gaining approval for the plant, AFWL has begun to gather some very preliminary information on the process and scope of effort required for DOD to build and operate a reactor plant. A recent report by the NUS Corporation (Pike and O'Reilly, 1982) for DOE has graphically illustrated the impact on plant construction if the DOD chose to let the Nuclear Regulatory Commission (NRC) license DOD plants. The incremental required safety effort in licensing the plant through the NRC could exceed 60 man-years and add years to the plant availability date. The NUS Corporation feels that DOD would be exempt from the NRC process under Section 91 of the Atomic Energy Act. I feel a more reasonable approval process for the Air Force would be modeled on the naval reactor program, in which the Navy provides safety analysis reports and asks for the NRC's advice. The NRC review in that case is advisory in nature and carries no mandatory provisions (Pike and O'Reilly, 1982). This process avoids the problem of public meetings on national security systems.

Another model may be the Interagency Nuclear Safety Review Panel, which was established to review for the purpose of approval by the President of the use of nuclear materials in the aerospace environment. This process is summarized in an excellent recent article by Dr. Gary Bennett (1981) of DOE. A similar panel could be formed for terrestrial nuclear power requiring presidential approval for construction and operation. This entire area is not well defined for DOD procedurally and needs much attention if DOD is serious about terrestrial nuclear power.

In summary, we have studied compact reactor sources for a specialized DOD application, deep basing of the MX missile. We have concluded that both LWRs and HTGRs can meet the power requirements and probably meet the survivability requirements for the hostile environments of this system. Both types of reactors will probably be employed in a modular, redundant manner to enhance survivability of the power system. An enormous amount of work remains before DOD can decide to really build a reactor system. A major effort is needed to address reactor component survivability and approval procedures. Finally, these studies are by no means definitive but are illustrative of the analysis of a potential DOD terrestrial mission for two reactor concepts. I am confident we can make this technology work to meet any DOD requirements for enduring, survivable power.

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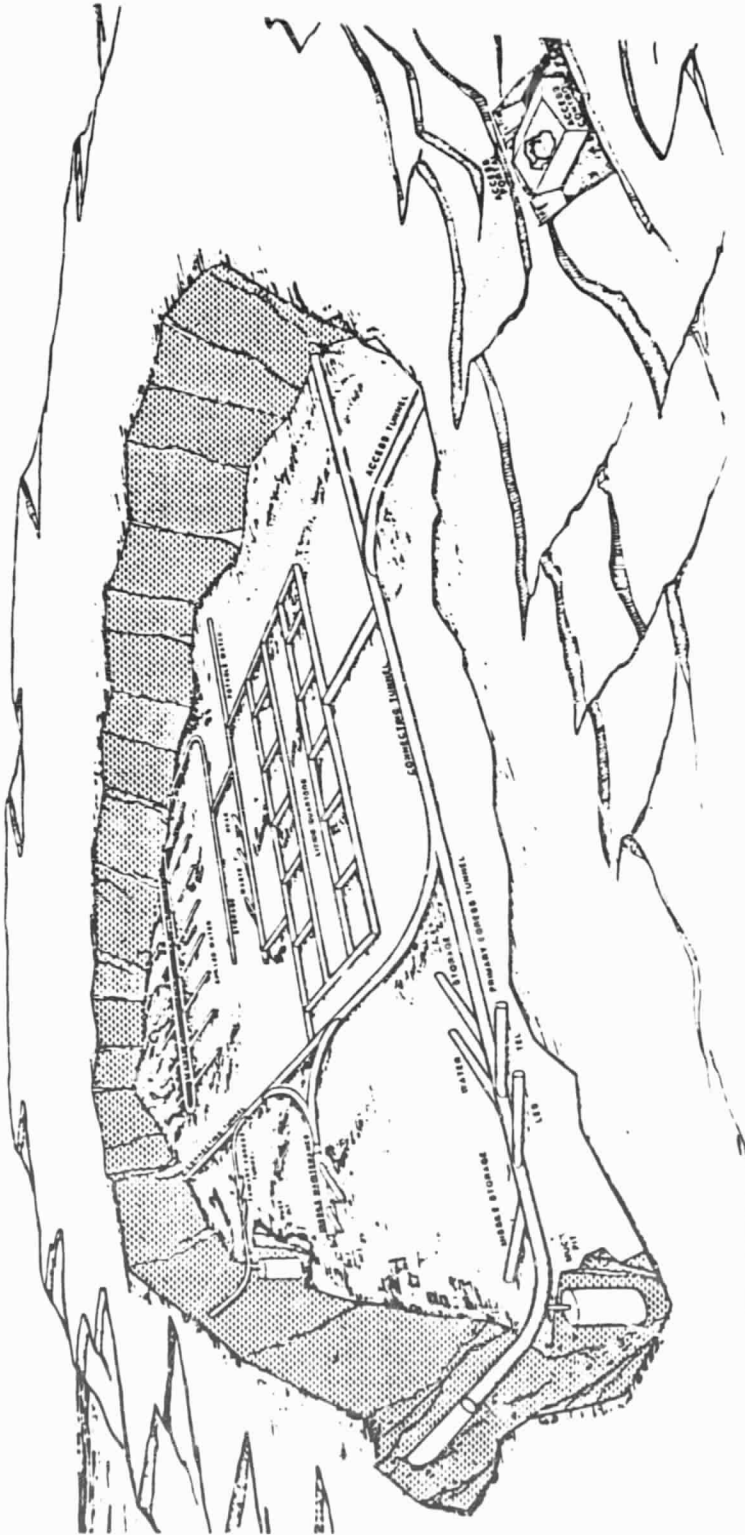
FIGURE 1   Interconnected inclined/clustered support facilities.
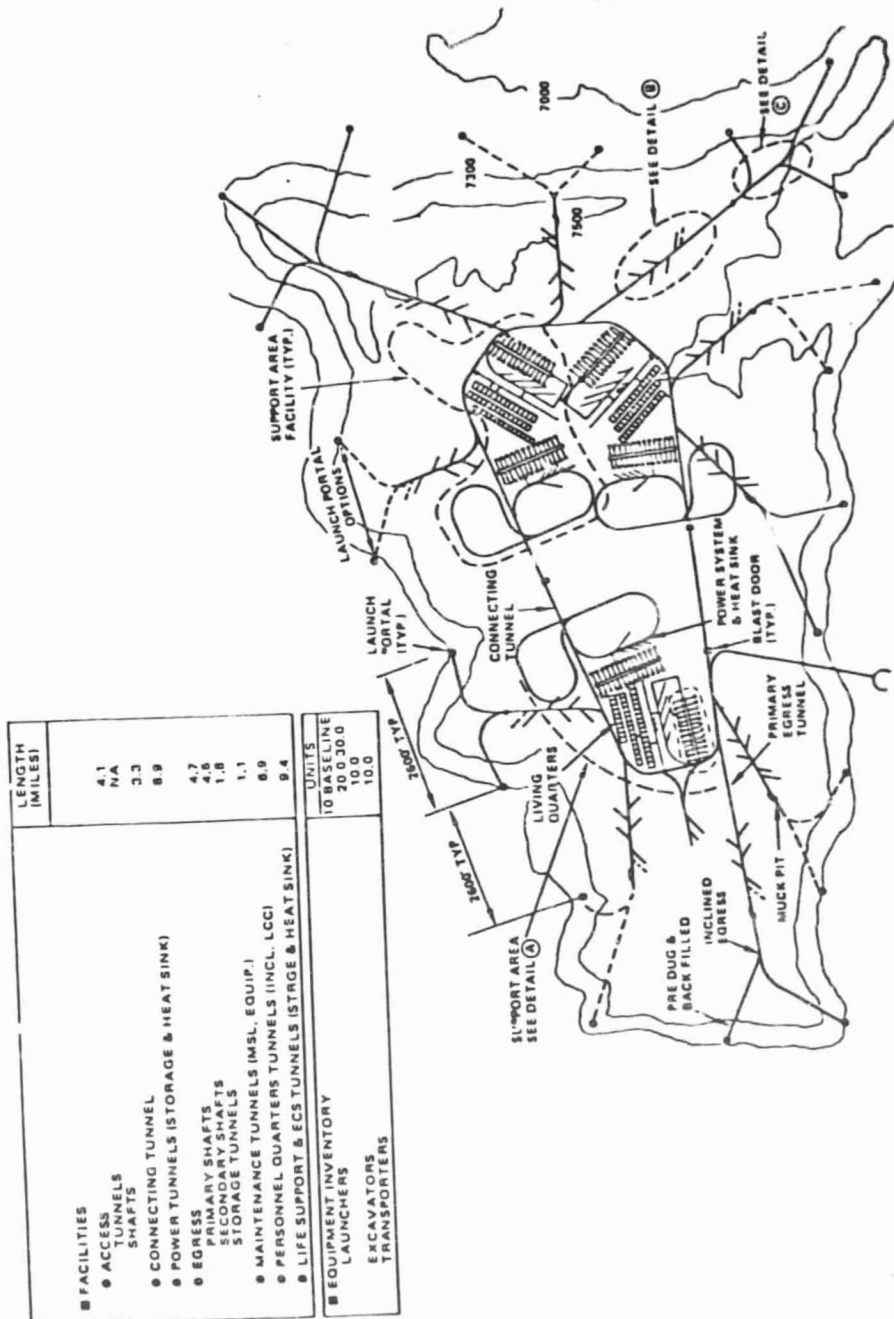
FIGURE 2 Inclined egress/clustered configuration.
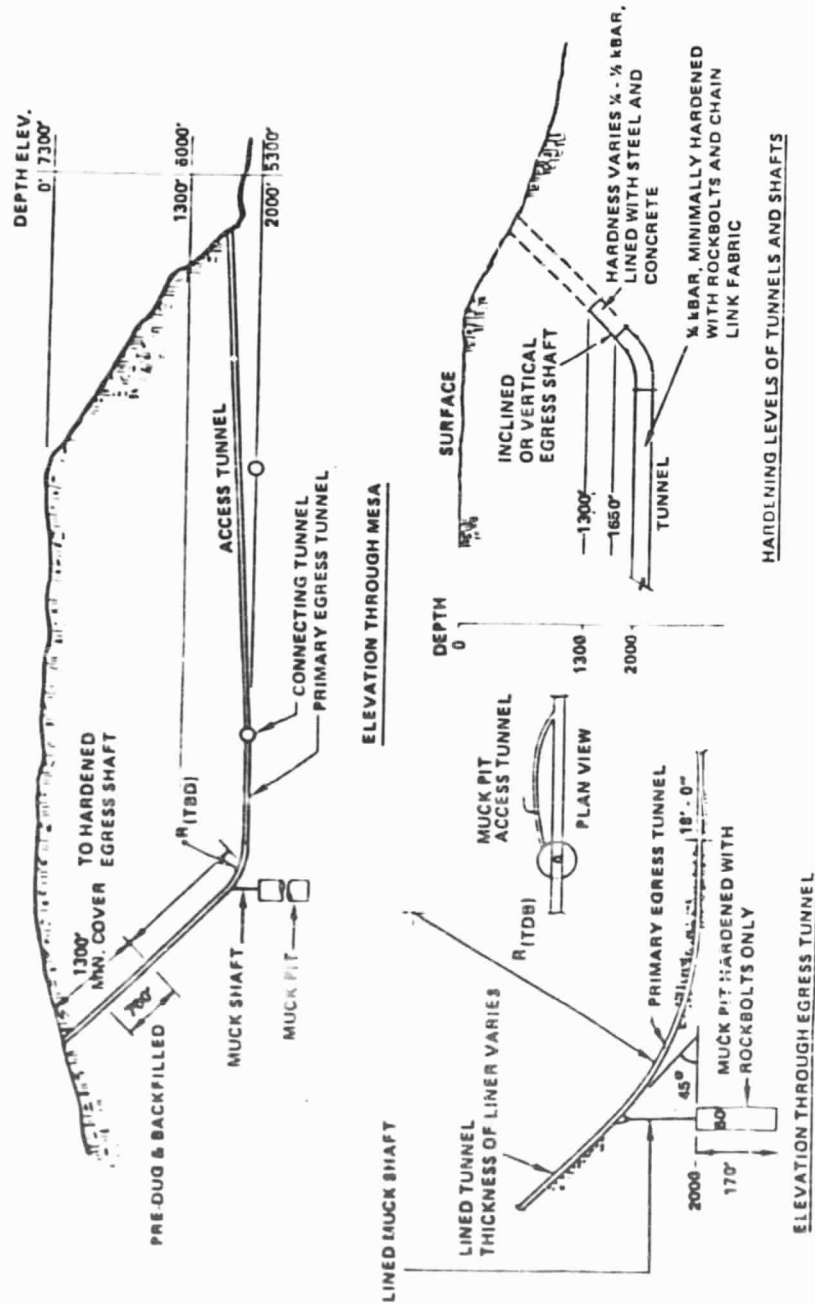
FIGURE 3 Inclined egress/clustered configuration: egress shaft detail.

PROPERTY	UNITS	LAYERS			
		Ⓐ	Ⓑ	Ⓒ	Ⓓ
UNCONFINED COMPRESSIVE STRENGTH	psi	9,100	1,300	2,500	20,000
TENSILE STRENGTH	psi	72	34	72	72
NATURAL BULK DENSITY	lb/ft³	147	112	118	165
GRAIN DENSITY	lb/ft³	156	156	150	168
POROSITY	%	9.3	35	37	2.8
WATER SATURATION (% OF PORE SPACE)	%	36	62	95.2	81
WEIGHT & WATER	%	5.2	12.3	19	.885
ACOUSTIC VELOCITY P-WAVE S-WAVE	ft/sec ft/sec	10,000 6,700	5,085 2,560	8,360 4,135	14,000 9,335

SIDE SLOPE TOP OF MESA 7300.0' 7000.0' SIDE SLOPE

APPROX. 300' 1000' 1500'

Ⓐ WELDED TUFF Ⓑ FRIABLE TUFF 8000.0' Ⓒ ZEOLITIC TUFF 4500.0'

Ⓓ WELDED TUFF & QUARTZITE

SECTION A-A

FIGURE 4   Characteristics of the study site.

CRATERS

64 MT (VOLUME ≈ 9.3 x 10⁸ FT³)
- LAYERED GEOLOGY

1 MT (VOLUME ≈ 4.0 x 10⁷ FT³)
- DRY SOFT ROCK

DEBRIS

64 MT AND 1 MT DIAGRAMS NOT TO SCALE
DIMENSIONS IN FEET

	64 MT				1 MT			
SURFACE OVERPRESSURE (KSI)	2	3	4	5	2	3	4	5
RANGE (KFT)	4.8	4.2	3.8	3.6	1.2	1.0	.96	.91
DEPTH (IN)	8	14	20	25	4	10	20	30
IMPACT VELOCITY (VERT COMPONENT) (FT/SEC)	280	260	250	240	140	130	125	120
BLANKET DENSITY (LB/FT³)		90				70		
PARTICLE DENSITY (LB/FT³)		150				110		

FIGURE 5 Craters and debris.
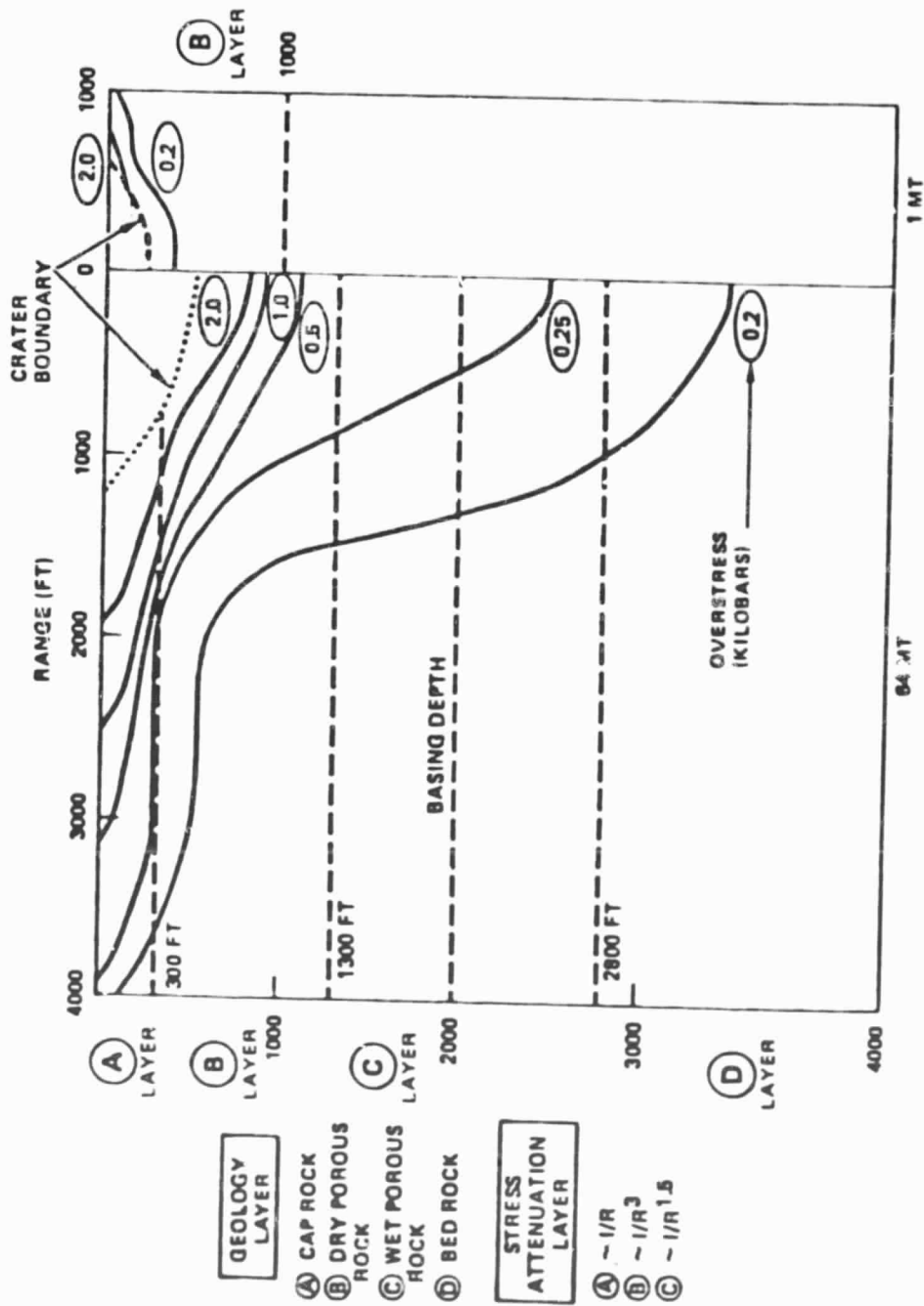
FIGURE 6   Peak principal overstress.
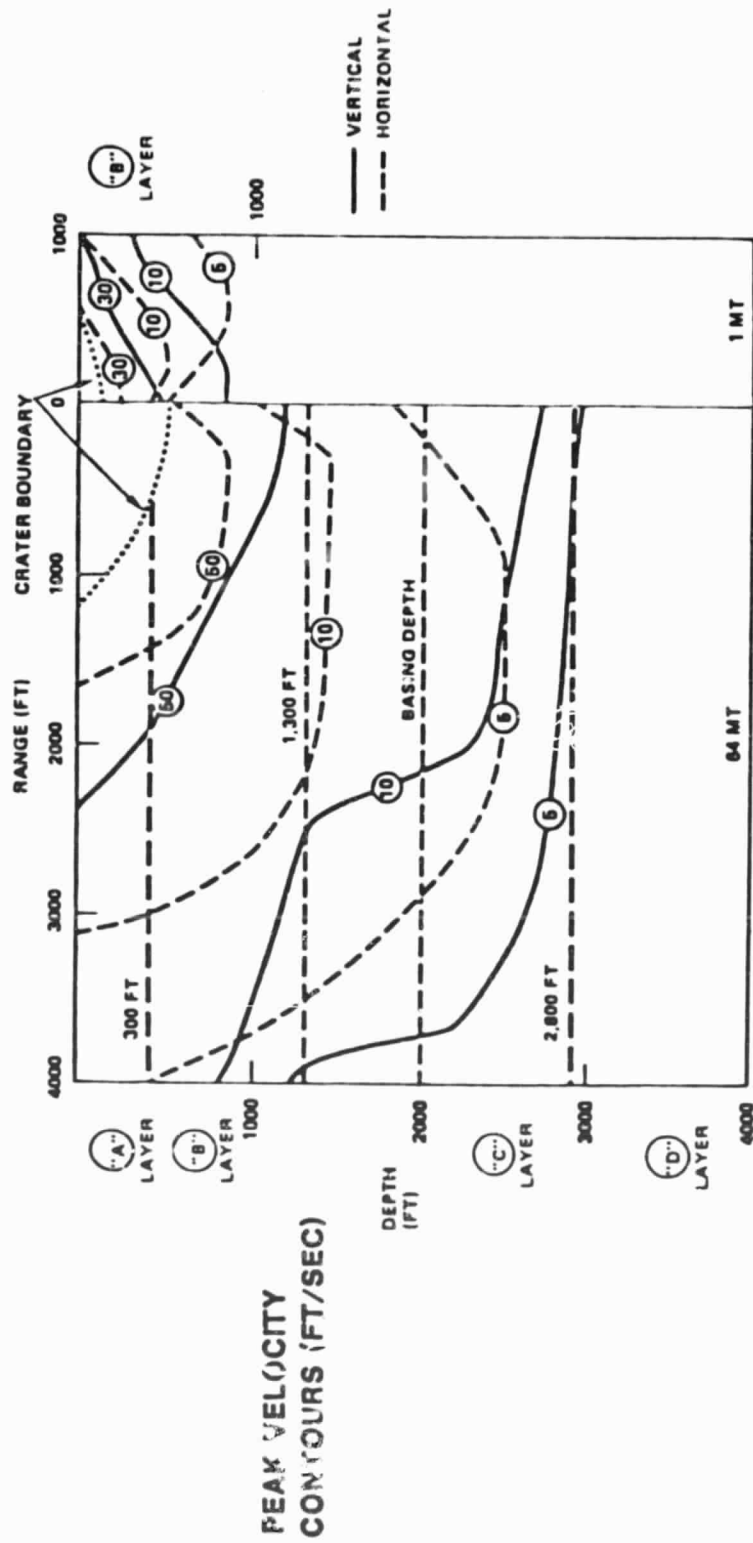
FIGURE 7 Velocities.

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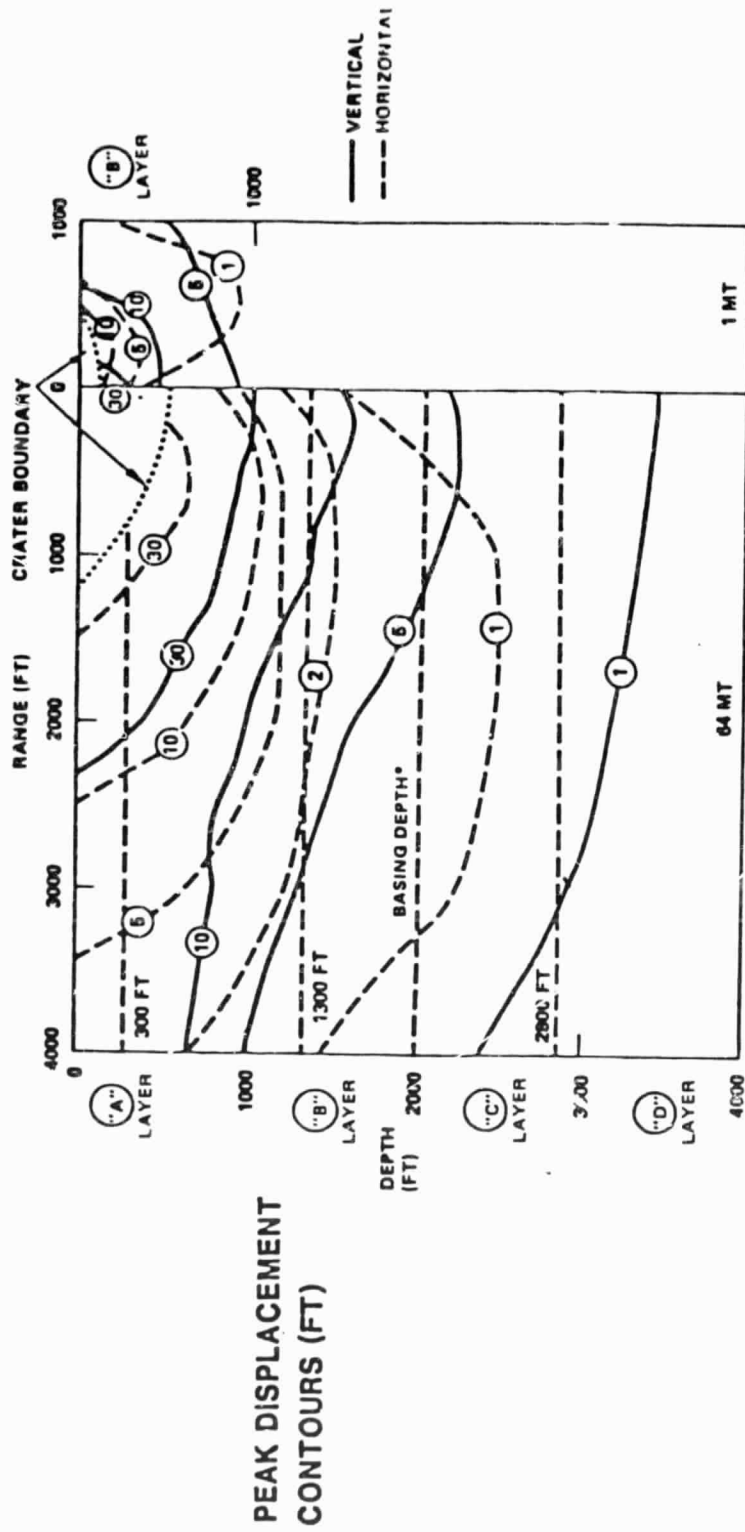
FIGURE 8 Displacements.

- PROVIDE PEACETIME POWER TO SURFACE AND DB FACILITIES

- POWER TO DB FOR DORMANT POST ATTACK PERIOD

- POWER FOR OPERATION OF EGRESS EQUIPMENT (TMB,MU┌K)

- POWER TO DB FOR OPERATION OF TEL EQUIPMENT

- EMERGENCY BACKUP POWER TO DB

FIGURE 9 Power requirements.

10 MISSILES

2 TUNNEL BORING MACHINES

2 SOIL MUCKING OPERATIONS

2 MISSILE LAUNCHERS

1 LAUNCH CONTROL-BUNKHOUSE SUPPORT
FACILITY

50 OPERATIONAL PERSONNEL

FIGURE 10   Composition of deep based MX missile squadron.

2.5KW PER STATIC MISSILE

750KW PER TBM

250KW PER MUCKING OPERATION

4.5MW PER MISSILE LAUNCHERS

100KW PER LAUNCH CONTROL

1KW PER PERSON FOR SUPPORT LOADS

750KW COMBINED COMMUNICATION EQUIPMENT
     COMPUTERS AND CONTROL LOADS

11.9MW*.45 = 5MW

FIGURE 11 Estimated power requirements.
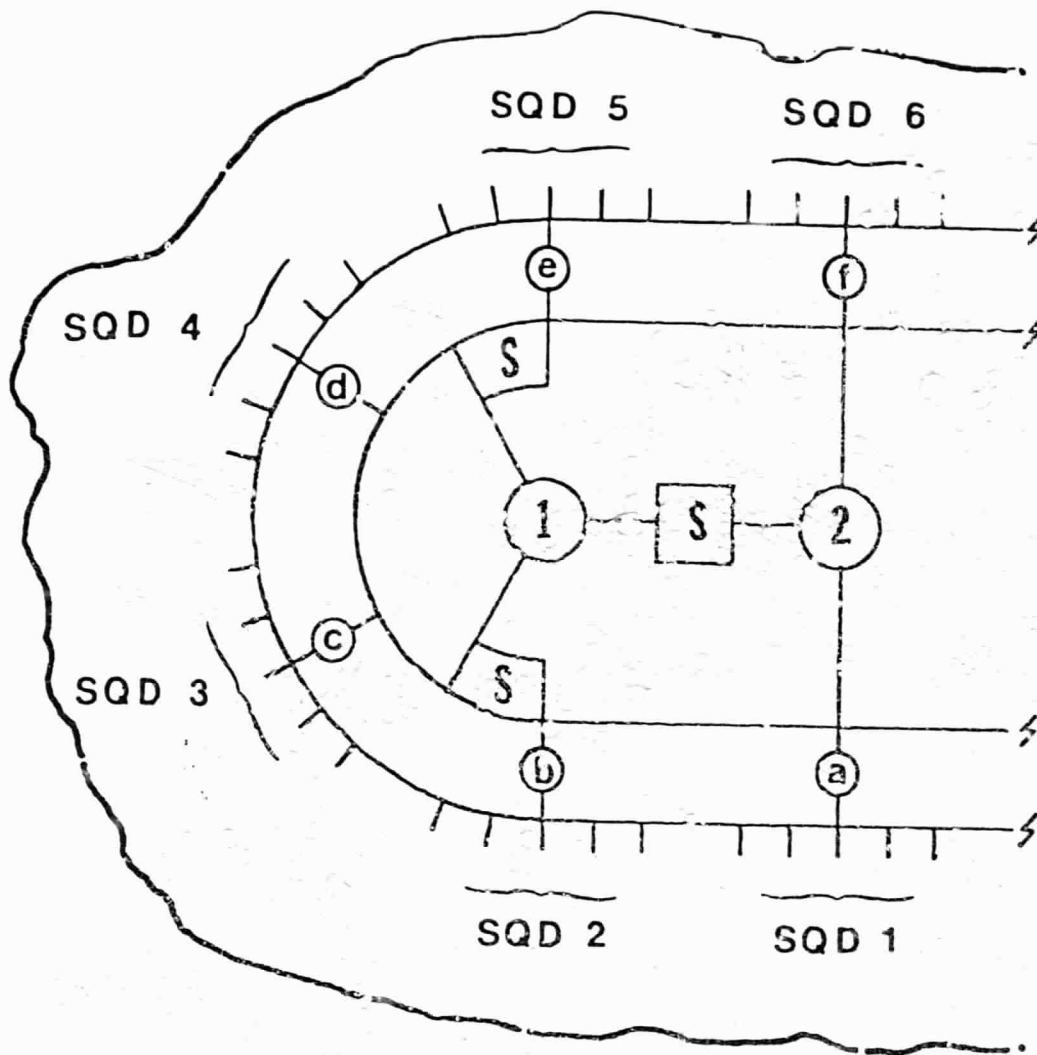
(1)   Prime power plants

(a)   Squadron collocated power plant

(S)   Mission support locations

FIGURE 12   Possible configuration of survivable power distribution system.
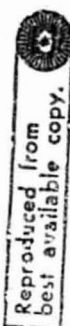
- FOUR + POWER CENTERS

- EACH CENTER HAS 2 EACH 28MWT REACTORS PRODUCING
  12.5MWe TOGETHER

- EACH CENTER HAS FIVE 4.5MWe TURBINE-GENERATORS

- CORE LIFE 28MWt+YR - CYCLE EFF =28%

- POWER SYSTEM COST $333M

- SURVIVABLE DUE TO MODULARITY

FIGURE 13 Features of power system using Army Package Power Reactor (APPR).
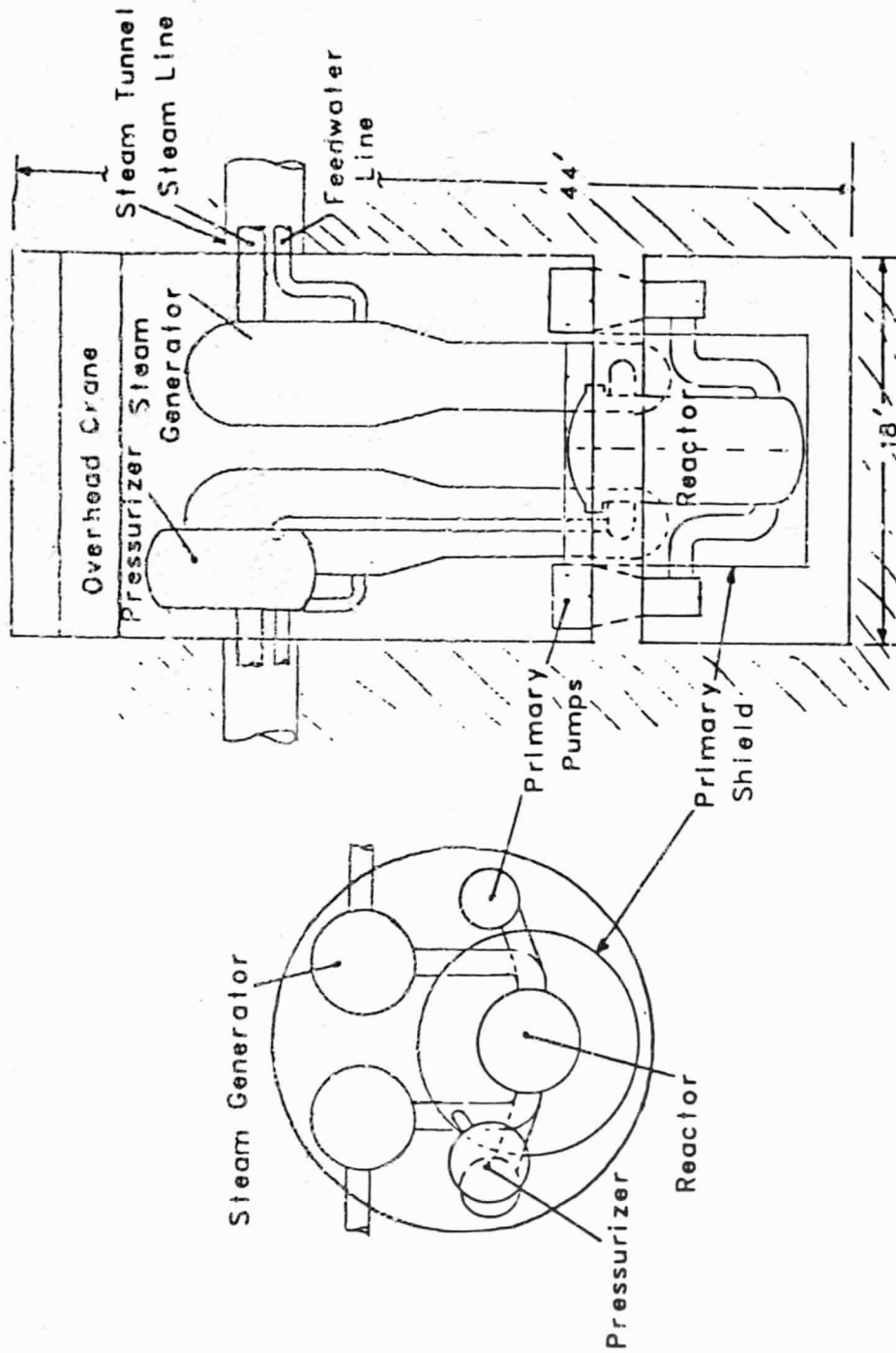
FIGURE 14   Reactor compartment layout for APPR.
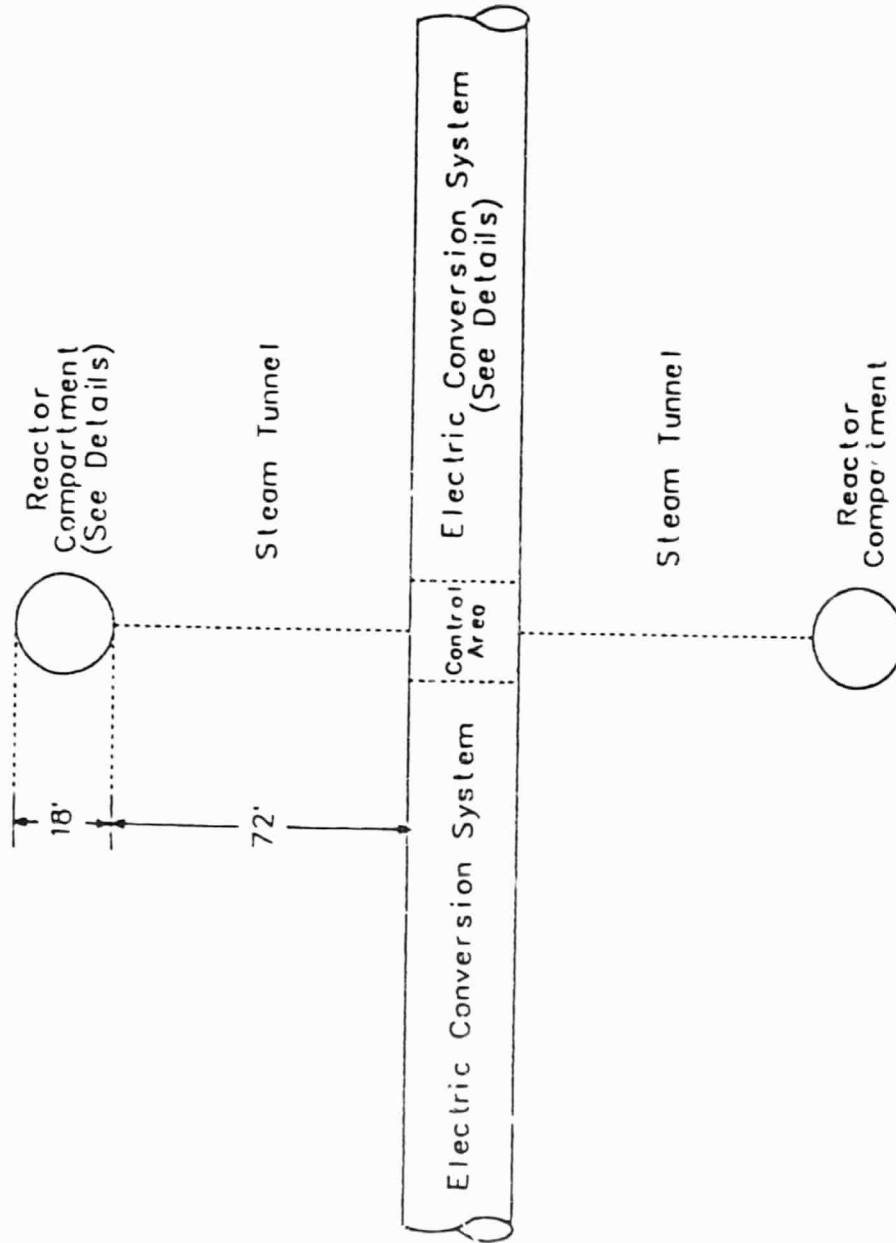
FIGURE 15 Conceptual layout of reactor and conversion system for APPR.

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-15MWe MODULES

-VERTICAL IN-LINE DESIGN (STEEL VESSEL)

-HEU PEBBLE FUEL (~6 CM DIAMETER)

-CYCLE EFF 28%

-WALK AWAY CAPABILITY

FIGURE 16   Features of High-Temperature Gas-Cooled Reactor (HTGR) for deep basing.
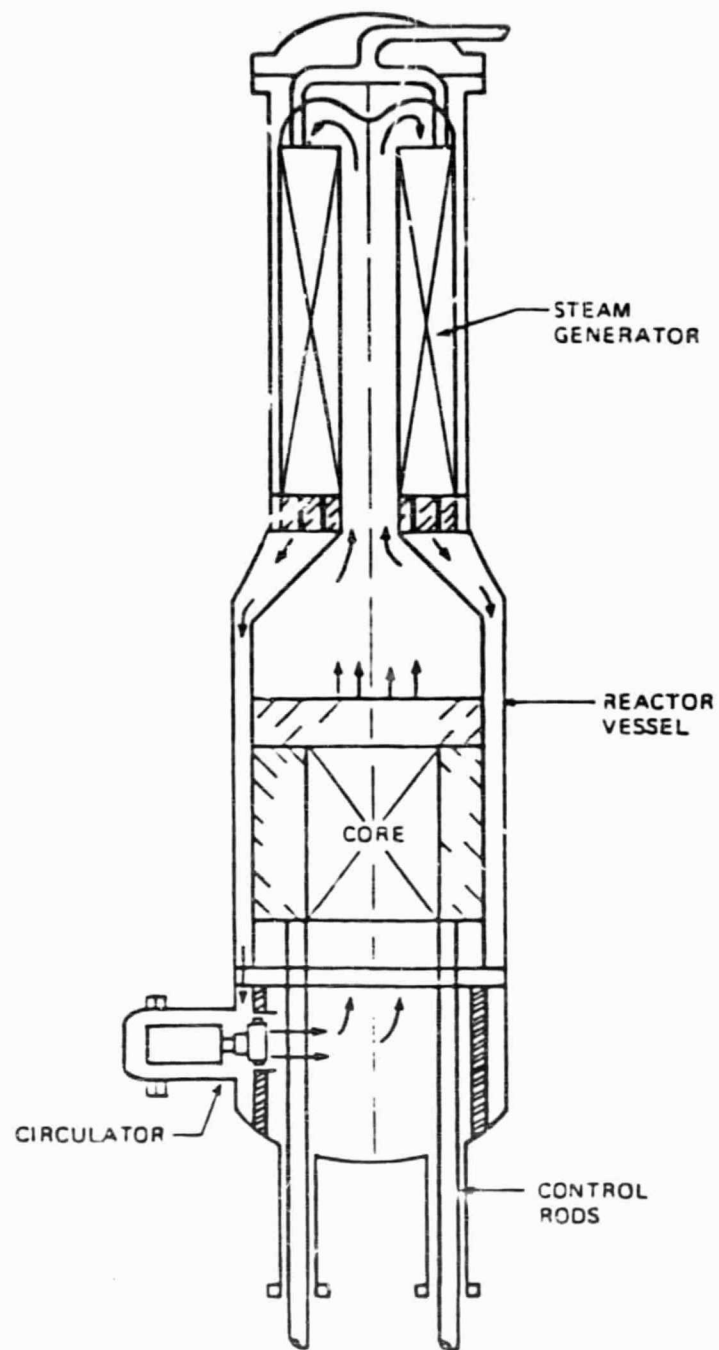
FIGURE 17   Conceptual design of HTGR primary system.
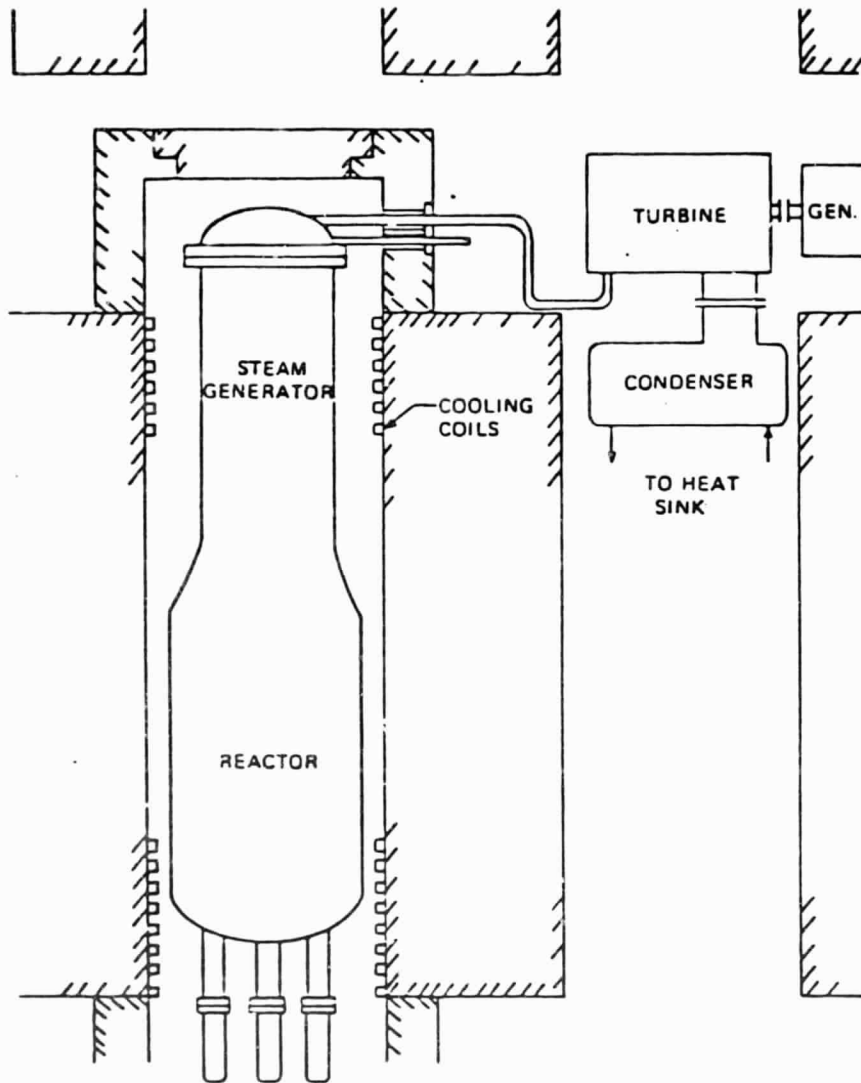
FIGURE 18 Preliminary HTGR system arrangement.

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POTENTIAL MISSION REQUIREMENTS
FOR NUCLEAR REACTORS AND ALTERNATIVES
IN SPACE AND PROPULSION APPLICATIONS

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OUTLOOK FOR SPACE NUCLEAR POWER DEVELOPMENT

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INTRODUCTION

I am pleased to open this second session, which focuses on the
potential mission requirements for nuclear reactors and alternatives
in space power and propulsion applications. However, before we look
further at mission requirements, it seems appropriate to consider the
outlook for space nuclear power development, which is my topic this
morning. In my formal paper, I describe in some detail the
background, current status, and future thrust of the space nuclear
power development program. (See the appendix following this paper.)
This morning, of course, I will only be able to briefly review the
major areas of the paper.

OVERVIEW

As shown in Figure 1, a large number of nuclear power systems have
been successfully used in space over the past three decades. There
have been 23 missions in which the Department of Defense (DOD) or the
National Aeronautics and Space Administration (NASA) has used nuclear
power supplies furnished by DOE or its predecessors. Clearly, DOE has
been, and continues to be, the government organization responsible for
space nuclear power development. We are very proud of our record in
providing reliable and safe space nuclear hardware to meet both
military and civilian mission goals in a timely manner.

A summary of these nuclear missions is shown in Table 1. The
history of these developments is interesting in its own right. The
engineering developments during and following World War II gave rise
to great advances in three new areas of technology: (1) electronics,
(2) rocketry, and (3) nuclear power. Studies in the late 1940s, such
as the Air Force's Project Feedback, suggested the merger of these
technologies for nuclear-powered spacecraft. The Atomic Energy
Commission (AEC), a predecessor of DOE, initiated a series of studies
in 1951 specifically to address the feasibility of powering spacecraft
with nuclear devices. Parallel studies were conducted on the

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feasibility of using nuclear power for the propulsion system. Out of
this study phase came several joint AEC/NASA and AEC/DOD programs to
investigate the use of nuclear systems for power and/or propulsion of
spacecraft. The government has long recognized the need for having a
nuclear power capability in space. This encouragement of nuclear
research and development was codified in the Atomic Energy Acts of
1946 and 1954, as amended, which assigned to the AEC the
responsibility to conduct, assist, and foster a program of research
and development to encourage the widespread participation in the
development and use of nuclear energy. In this way, Congress and the
President focused the nation's efforts in the nuclear area, defined
the separation of powers, and ensured the safe control of the special
nuclear material needed for these research and development programs.
This concentration of nuclear responsibility continues with the Energy
Reorganization Act of 1974 and the Department of Energy Organization
Act of 1977.

From this farsighted planning by the Congress and the executive
branch, DOE and its predecessor agencies have been able to provide
NASA and DOD with all of their space nuclear power supply needs. In
carrying out its charter, a large number of programs have been
conducted by DOE to satisfy various power ranges. Looking to the
future, it is apparent that a broad range of power outputs and
lifetimes will need to be considered for the growing spectrum of
potential mission needs, which are depicted in Figure 2.

The functional missions flown to date have required low power and
long life, and these needs have been met by using radioisotope
thermoelectric generators, or RTGs. As future power needs rise to
1 kW(e) and above, the higher conversion efficiencies of dynamic
electric conversion systems will become increasingly attractive,
making them the preferred power system. This is apparent when we
recognize that conversion efficiencies of 15-25 percent are possible
with dynamic systems compared to only 5-10 percent for thermoelectric
systems. In an effort to capitalize on this advantage, a ground
prototype system of the Dynamic Isotope Power System (DIPS) has been
built and operated for 5,000 hours. At a power level of about 10
kW(e), the increasing inventory of fuel needed in the DIPS will make
reactor concepts the preferred system. At 25-kW(e) and above, reactor
systems become the most attractive in cost, mass, and size among all
competing power systems.

Figure 3 depicts the various heat source types and conversion
system options for the power ranges graphically shown in Figure 3. I
will be discussing many of these with you this morning, beginning with
the RTGs.

Radioisotope Thermoelectric Generators

The RTG power system has been a cornerstone of the space nuclear
program. As I am sure you know, RTGs were used by our astronauts on

TABLE 1   Summary of Space Nuclear Power Systems Launched by the United States (1961-1980)

Power Source[a]	Spacecraft	Mission Type	Launch Date	Status
SNAP-3A	Transit 4A	Navigational	June 29, 1961	Successfully achieved orbit
SNAP-3A	Transit 4B	Navigational	Nov. 15, 1961	Successfully achieved orbit
SNAP-9A	Transit-5BN-1	Navigational	Sept. 28, 1963	Successfully achieved orbit
SNAP-9A	Transit-5BN-2	Navigational	Dec. 5, 1963	Successfully achieved orbit
SNAP-9A	Transit-5BN-3	Navigational	April 21, 1964	Mission aborted; burned up on reentry
SNAP-10A (reactor)	SNAPSHOT	Experimental	April 3, 1965	Successfully achieved orbit
SNAP-19B2	Nimbus III	Meteorological	April 14, 1969	Successfully achieved orbit
SNAP-27	Apollo 12	Lunar	Nov. 14, 1969	Successfully placed on lunar surface
SNAP-27	Apollo 13	Lunar	April 11, 1970	Mission aborted on way to moon. Heat source returned to South Pacific Ocean
SNAP-27	Apollo 14	Lunar	Jan. 31, 1971	Successfully placed on lunar surface
SNAP-27	Apollo 15	Lunar	July 26, 1971	Successfully placed on lunar surface
SNAP-19	Pioneer 10	Planetary	March 2, 1972	Successfully operated to Jupiter and beyond
SNAP-27	Apollo 16	Lunar	April 16, 1972	Successfully placed on lunar surface
Transit-RTG	Transit (TRIAD-01-1X)	Navigational	Sept. 2, 1972	Successfully achieved orbit
SNAP-27	Apollo 17	Lunar	Dec. 7, 1972	Successfully placed on lunar surface
SNAP-19	Pioneer 11	Planetary	April 5, 1973	Successfully operated to Jupiter and Saturn and beyond
SNAP-19	Viking 1	Mars	Aug. 20, 1975	Successfuly landed on Mars
SNAP-19	Viking 2	Mars	Sept. 9, 1975	Successfully landed on Mars
MHW	LES 8	Communications	March 14, 1976	Successfully achieved orbit
MHW	LES 9	Communications	March 14, 1976	Successfully achieved orbit
MHW	Voyager 2	Planetary	Aug. 20, 1977	Successfully operated to Jupiter and Saturn and beyond
MHW	Voyager 1	Planetary	Sept. 5, 1977	Successfully operated to Jupiter and Saturn and beyond

[a] SNAP stands for Systems for Nuclear Auxiliary Power. All odd-numbered SNAP power plants use radioisotope fuel. Even-numbered SNAP power plants have nuclear fission reactors as a source of heat. MHW stands for the Multihundred Watt RTG. LES stands for Lincoln Experimental Satellite.

SOURCE: G. Bennett, Overview of the U.S. Flight Safety Program, Nucl. Safety, 22(4), 424, 1981.

the lunar surface, with excellent results. As illustrated in Figure 4, they have also been used by NASA in various interplanetary missions, including those to Jupiter, Saturn, and the Martian surface, and they have been used by DOD for military applications, like LES 8/9. These programs, as well as the current RTG programs for NASA's Galileo and Solar Polar missions illustrated in Figures 5 and 6, have given DOE a great deal of experience in maintaining the technical and programmatic interfaces with a user agency.

The isotope thermoelectric generator program continues to progress, as illustrated in Figure 7. Under development are the nuclear power units to be launched aboard the NASA Space Shuttle in 1986 for the Galileo and the International Solar Polar missions.

Figure 8 shows the progress made in the isotope thermoelectric power systems and the objectives DOE has set for future development. As you can see, there is a steady progression in power, efficiency, and specific power. The latest generation of RTG technology development for potential use in U.S. space missions beginning in the 1988 time frame is the modular isotope thermoelectric generator (MITG). It has been estimated that about $45 million and 4 years would be required to complete the research and technology verification phase of this advanced RTG system featuring modularity and greatly increased specific power.

Dynamic Isotope Power Systems

To meet potential DOD future space missions forecast in the 1- to 2-kW(e) power range, DOE embarked on a development program to evaluate dynamic conversion systems that combine a radioisotope heat source with a rotating turbine/alternator system. This type of system, depicted in Figure 9, can yield efficiencies approaching 18-25 percent. To date, prototypic equipment has been operated in excess of 5,000 hours in a space vacuum. Approximately $20 million was spent during the course of this program. The flexibility of this type of space isotope power system in mating with the spacecraft is illustrated in Figure 10. The Nuclear Integrated Multimission Spacecraft (NIMS) can accommodate four major identified DOD mission categories, i.e., communications, surveillance, navigation, and meteorology. It has been estimated that approximately $40 million and about 3 years would be required to qualify this system for space flight.

Space Nuclear Reactor Power Systems

As we previously discussed, in order to meet higher mission power requirements, nuclear reactors would be required. From the mid-1950s to 1973, the AEC carried out an active space reactor power system development program that focused on some nine concepts, including

- o zirconium hydride reactors
- o liquid-metal-cooled reactors
- o nuclear rocket--Rover/NERVA
- o boiling metal reactors
- o gas-cooled reactors
- o thermionic reactors
- o dynamic and static power conversion.

For brevity, I will review only three of these, but a more detailed discussion can be found in my formal paper (see appendix).

I am sure that you have frequently heard of the SNAP-10A reactor that was launched in 1965--the only U.S. reactor to fly and the first in the world. But I am not sure if you are aware of the fact that a duplicate reactor was successfully ground tested for 10,000 hours. This program was part of a broader zirconium hydride reactor development effort, illustrated in Figure 11, which gives some perspective of the extent of this program.

Rover/NERVA

Perhaps the most challenging application of nuclear energy to spacecraft was the nuclear rocket program. Glenn T. Seaborg summed up the challenge as follows: "What we are attempting to make is a flyable compact reactor, not much bigger than an office desk, that will produce the power of Hoover Dam from a cold start in a matter of minutes."

The United States carried out an ambitious research program from 1955 to 1972 aimed at developing a capability to use nuclear power for rocket propulsion, and some 18 reactor systems were built and operated. This program, which was jointly managed by the Atomic Energy Commission/United States Air Force (AEC/USAF) and AEC/NASA, demonstrated a number of important firsts in space nuclear technology: (1) multiple-start capability, (2) peak power of 4,200 MW, (3) 60-min continuous operation, sufficient for many space missions, at a gigawatt power level, and (4) ability to start on its own power and operate stably over a wide range of conditions. From this research evolved the Nuclear Engine for Rocket Vehicle Applications, or NERVA, that was designed for use on a reusable nuclear shuttle. NERVA was to produce 75,000 lb of thrust at a specific impulse of 825 s--twice the specific impulse of the best chemical rockets. Such a capability translates into larger payloads and shorter flight times. Furthermore, NERVA offered the possibility of producing 15-25 kW of power for hotel functions during the coast phase of the mission.

The Rover/NERVA program demonstrated again that several government agencies can cooperatively work together to apply nuclear technology to spacecraft and that nuclear power can provide a quantum leap in performance improvement over more conventional systems.

Thermionic Reactor

In a more exotic vein, the thermionic reactor was a concept for converting heat to electricity inside a reactor core using the thermionic process. It can cover the range from a few kilowatts to several megawatts of electric power, but the emphasis in the last several years of activity was on a 120-kW system suitable for space propulsion missions.

The AEC program began in fiscal year (FY) 1959 with support of the Los Alamos Scientific Laboratory "plasma thermocouple." The AEC and NASA thermionic efforts were administratively combined in FY 1971, with NASA assuming the responsibility for systems studies and support technology.

All U.S. space reactor development programs were terminated in 1973 because of budgetary pressures and changes in user interest. Subsequently, a DOD/Energy Research and Development Administration (ERDA) Space Nuclear Application Steering Group was organized in 1975 with the objective of establishing and maintaining the necessary management interface and communication channels between DOD and ERDA for the purpose of ensuring the effective and efficient military use of nuclear energy for space and other directly related applications. In concert with the steering group, an Advanced Space Power Working Group was established and provided a report in 1977. This report assessed DOD potential missions and the various space power technologies. The report formed the basis for DOE to enter into a 5-year space reactor technology program beginning in 1979 at approximately $2 million/yr. At that time, DOD space system power requirements appeared to be in the 10- to 100-kW(e) range. The technology development program focused on a reactor system concept called Space Power Advanced Reactor (SPAR).

In 1981, while DOD decided to continue to study its mission applications and could offer no direct support, the NASA mission models indicated that 100 kW(e) could provide a suitable power level for both outer planetary and earth orbital missions. The work associated with the power conversion subsystem was planned to be conducted by NASA, while the efforts on the reactor subsystem would continue to be conducted by DOE with NASA support. In FY 1982, with NASA support, the reactor development program was tailored to NASA objectives that identified a number of mission considerations for the 100-kW(e) power system, and the effort was redesignated the Space Nuclear Reactor Power System Technology Program (SP-100).

Since the summer of 1981 we have experienced a renewed interested by DOD in both the kilowatt and the megawatt power ranges. Some of the reasons for the renewed military interest are shown in Figure 12. As you can see, nuclear power enhances survivability against nuclear attack, laser attack, and antisatellite attack. It also makes it practical to provide the payload with high power, which enhances survivability by permitting higher orbits, more ground links, harder electronics, smaller antennas, and mobile ground receivers. Nuclear

power also provides the spacecraft with an improved field of view and improved pointing accuracy and permits undegraded operation in the Van Allen radiation belts.

To meet the growing user interest, DOE has established an Office of Space Reactor Projects, as shown in Figure 13. As you can see, this office will report directly to me, and through me to the assistant secretary for nuclear energy. The Office of Space Reactor Projects will be responsible for the activities listed in Figure 14. The near-term efforts of the office will focus on a review of past and present space reactor deve' 'ment activities. A variety of technologies will be examined to ensure that the reactor development efforts to follow will emphasize those concepts with the highest possible likelihood of successful mission performance. This review effort will then form the basis of a space reactor technology plan that will be coordinated and agreed upon with user agencies and will be updated on a regular basis. I want to emphasize that this office will be responsible only for the reactor development!

In the context of the reevaluation process, it is important that an extensive review be made of the various reactor technology options, listed in Figure 15. As part of this process, Rockwell International is under contract to DOE to conduct a two-phase program, which is to be completed in about 6 months. Phase I activities include the coll   r   and assessment of space nuclear reactor power systems information. Phase II efforts would include the collection and validation of mission requirements, along with the synthesis of the power systems analysis data, and culminate in a development schedule and recommended program plan acceptable to the user agency.

The present program at Los Alamos National Laboratory (LANL) is focused on the Heat Pipe Reactor (Figure 16). The reactor core is cooled by 120 lithium heat pipes with Mo-Re walls and fins. The spaces between the fins are filled with UO_2 fuel wafers. Figure 17 illustrates the overall heat pipe reactor (HPR) power system. The core heat pipes extend past the shield through the whole length of the power system. The generated he   transferred by radiation from the central pipe bundle to the thermoelectric converters mounted on the radiator panels. Radiative coupling eliminates most connections between the reactor system and the conversion system and facilitates technical and organizational separation of the two development programs. The key features of the HPR design are summarized in Figure 18, and Dr. Boudreau from LANL will describe this program in some detail later in the conference.

It is planned to continue this HPR development program, and if adequate funds are forthcoming from NASA, current plans call for completion of the "critical technology development phase" by the end of FY 1986, at which time a key decision will be made whether to proceed into a "ground engineering system phase" for the HPR system.

Mission analyses, only now beginning in earnest by user agencies, may direct us away from the HPR concept or may result in parallel

development of more than one reactor concept. What is important is that we maintain program integrity while missions are being defined.

As I noted previously, one of the early activities of the Office of Space Reactor Projects will be to reevaluate past and present space reactor activities for relevance to future needs. This review will focus on those technologies that offer a contribution to future high-technology systems. For example, the ZrH reactor systems are limited in temperature and power density capabilities and do not appear to offer much in reactor core technology for the future. Many reactor systems appear to show more promise, and we will be actively evaluating rotating- and fixed-bed reactors, high-temperature gas reactors, liquid-metal systems, fluid-fuel systems, as well as a range of power conversion technologies.

In order that DOE might carry out its function to develop and deliver the space power systems needed in the future, it is extremely important that detailed and thorough mission assessments be accomplished in a timely manner. In addition, it is also important that the nuclear reactor power system technologies assessment be carried out to permit the necessary trade-offs between the mission studies and the analyses of spacecraft systems and power supply subsystems. This process is generalized in Figure 19. Only by using these trade-off studies can one select the appropriate nuclear reactor power system technologies for development to yield a high probability of meeting the user agencies' needs. It is clear that a space nuclear power capability will require the consideration of many technical alternatives, and what we need is a clear definition of missions and power needs.

As summarized in Figure 20, DOE has been charged with the responsibility of space reactor development, and we have had years of experience with the development of many specialized reactors prototypical of space nuclear power systems. This work has been accomplished through the efforts of many people--managers, engineers, craftsmen, technicians--in the national laboratories and in private industry. National laboratories that have been particularly active in the field of space nuclear power include Los Alamos, Oak Ridge, Savannah River, Brookhaven, Livermore, Argonne, Sandia, Hanford, Idaho National Engineering Laboratory, the Mound Facility, and the Applied Physics Laboratory. Industrial participants range from very large to quite small and include General Electric, Westinghouse, Rockwell International, Aerojet, General Atomic, TRW, Thermo Electron, Teledyne Energy Systems, NUS Corporation, Fairchild Industries, and many others. These widespread activities have created over the years a large and diversified population of technical experts with interest and experience in advanced nuclear power systems. This resource is waiting to be reengaged and redirected in a new effort to put nuclear power to work in space.

- US active on studies and development of Space Nuclear Power and Propulsion since 1951.

- 38 Nuclear Power Systems successfully flown on 23 DOD and NASA missions.

- All but one system were RTGs, Radioisotope Thermoelectric Generator.

- Unlike USSR, the US has only flown one reactor in space, the 500-watt SNAP-10A in 1965.

- Many other, higher-powered systems under development by DOE were terminated short of flight tests, because of budgetary pressures and/or changes in user interest.

- Pursuant to the Atomic Energy Acts of 1946 and 1954 and the Energy Acts of 1974 and 1977, the above programs were conducted by DOE and its predecessors, in cooperation with the DOD and NASA user agencies.

- With current renewed interest, DOE stands ready to cooperate with DOD and NASA to continue supplying them with all of their space nuclear power supply needs.

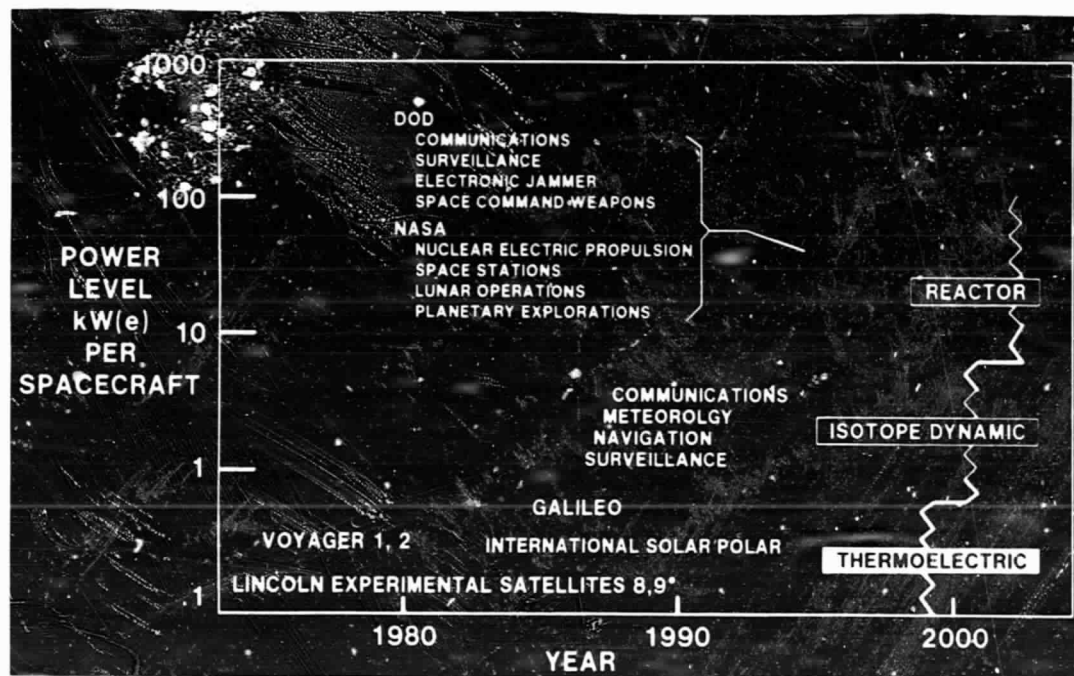FIGURE 1   Summary of U.S. space nuclear power development (1951-present).

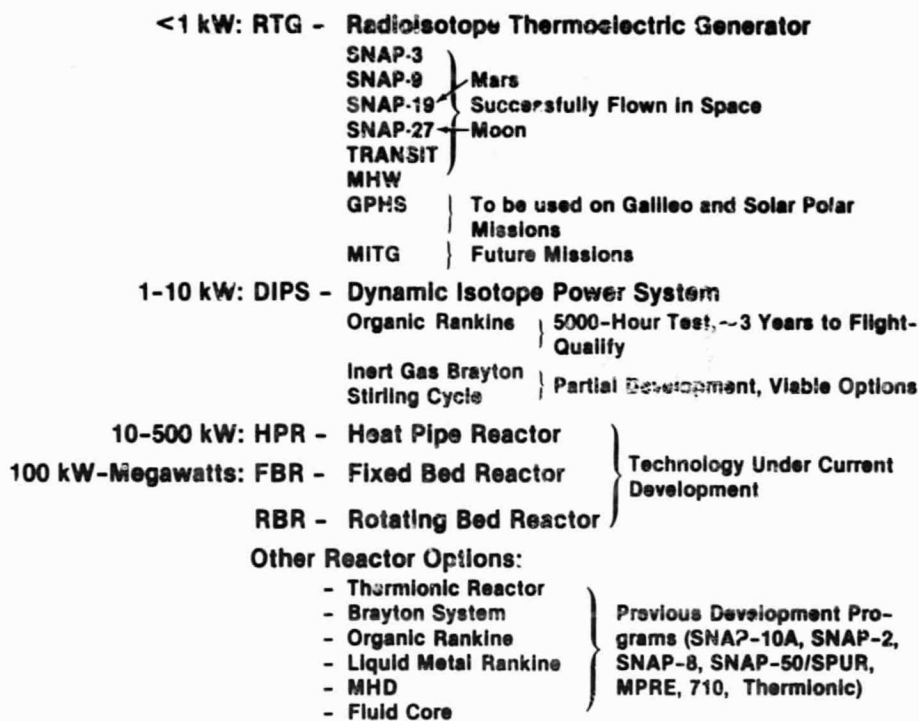FIGURE 2   Space nuclear power and evolving mission requirements.

<1 kW: RTG – **Radioisotope Thermoelectric Generator**

> SNAP-3
> SNAP-9 } Mars
> SNAP-19 { Successfully Flown in Space
> SNAP-27 —— Moon
> TRANSIT
> MHW
> GPHS | To be used on Galileo and Solar Polar
> | Missions
> MITG   } Future Missions

1-10 kW: DIPS – **Dynamic Isotope Power System**

> Organic Rankine   } 5000-Hour Test, ~3 Years to Flight-
>   Qualify
>
> Inert Gas Brayton } Partial Development, Viable Options
> Stirling Cycle

10-500 kW: HPR – **Heat Pipe Reactor**

100 kW-Megawatts: FBR – **Fixed Bed Reactor**   } Technology Under Current
                                                          Development
RBR – **Rotating Bed Reactor**

Other Reactor Options:

> – Thermionic Reactor
> – Brayton System   } Previous Development Pro-
> – Organic Rankine   grams (SNAP-10A, SNAP-2,
> – Liquid Metal Rankine   SNAP-8, SNAP-50/SPUR,
> – MHD   MPRE, 710, Thermionic)
> – Fluid Core

FIGURE 3 Space nuclear power options (for the power ranges shown in Figure 2).

LES 8/9 NEAR SYNCHRONOUS EARTH
ORBITING COMMUNICATIONS SATELLITE

PIONEER – 2 JUPITER ENCOUNTERS
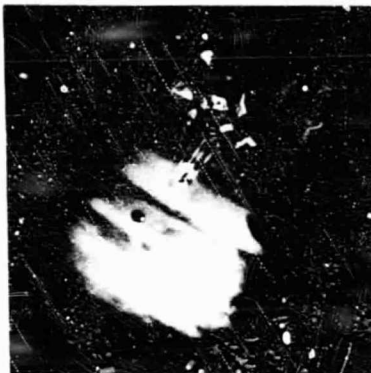
- 2ND VEHICLE ENCOUNTERED BOTH JUPITER & SATURN
- COLLECTION OF ENVIRONMENT DATA BASE
- RTG POWER SUPPLIES AVERAGE POWER >130 WATTS
- POWER STILL OPERATING AFTER 8 & 9 YEARS

- 2 SUCCESSFUL FLIGHTS
- 5 YEARS IN OPERATION. AVERAGE POWER >140 WATTS

VOYAGER - 2 JUPITER & SATURN ENCOUNTERS

VIKING - 2 MARS LANDERS

- 1ST MARS SCIENTIFIC EXPLORATION FOR EXTRATERRESTRIAL LIFE
- 1 POWER SUPPLY STILL OPERATING AFTER 5 YEARS
- RTG POWER SUPPLIES AVERAGE POWER LEVEL >70 WATTS

- REVEALED VALUABLE SCIENTIFIC DATA
- ONE FLIGHT CONTINUING TO URANUS IN 1986 & NEPTUNE IN 1989
- 4 YEARS IN OPERATION
- AVERAGE POWER >150 WATTS

FIGURE 4   Thermoelectric application for earth orbit and planetary
missions.

FIGURE 5   Thermoelectric applications: Galileo mission.

INTERNATIONAL
SOLAR POLAR
MISSION

JOINT NASA-ESA EFFORT
FIRST OUT OF THE ECLIPTIC SOLAR MISSION
FIRST EXPLORATION OF THE SUN'S POLAR REGION
LAUNCH 1986.
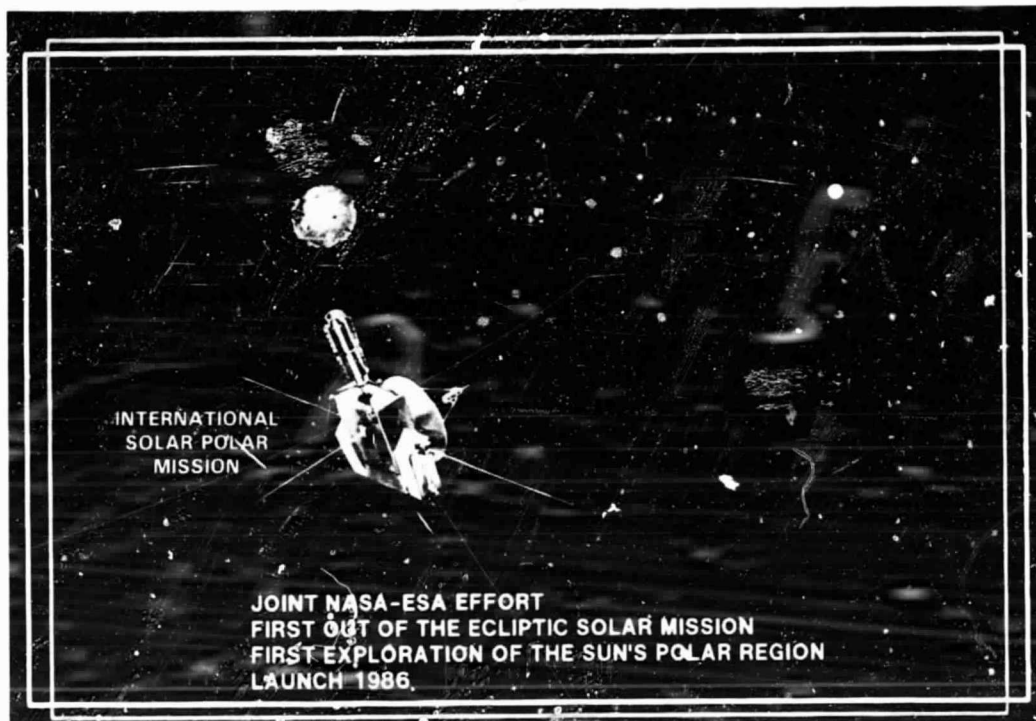
FIGURE 6   Thermoelectric systems: International Solar Polar mission.
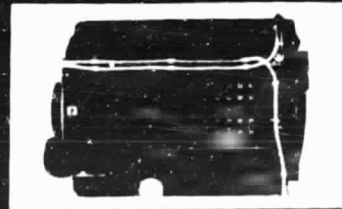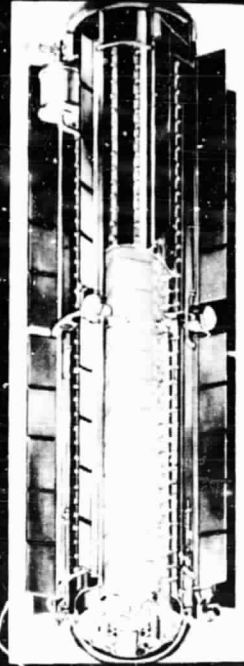
FIGURE 7   Progress in the radioisotope thermal generator program.

Generator	SNAP-3	SNAP-27	SNAP-19	MHW	GPHS-RTG	MITG
Mission	Transit-4	Apollo	Pioneer & Viking	LES 8/9 & Voyager	Galileo & Solar-Polar	future
Power-watts	2.5	65	40	150	285	~300
Fuel Form	Pu metal	PuO_2 micro-spheres	PuO_2-Mo cermet	dense PuO_2	dense PuO_2	dense PuO_2
Fuel Clad	Ta	Super-alloy	T-111	Ir	Ir	Ir
Thermo-electrics	PbTe	PbSnTe	PbTe-TAGS	SiGe	SiGe	SiGe or SiGe/GaP
Conversion Efficiency	5.0	5.0	6.3	6.7	6.8	8.2
Specific Power, watt/kg	1.48	2.34	3.00	3.94	5.14	9.05

FIGURE 8   Trends in RTG technology.
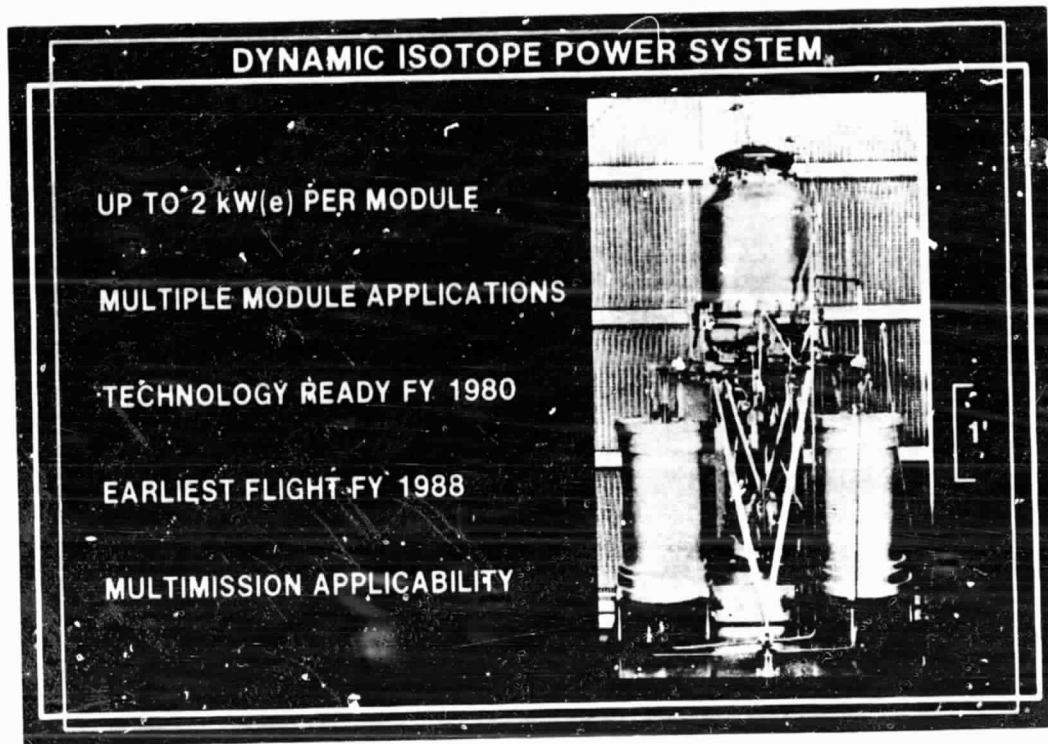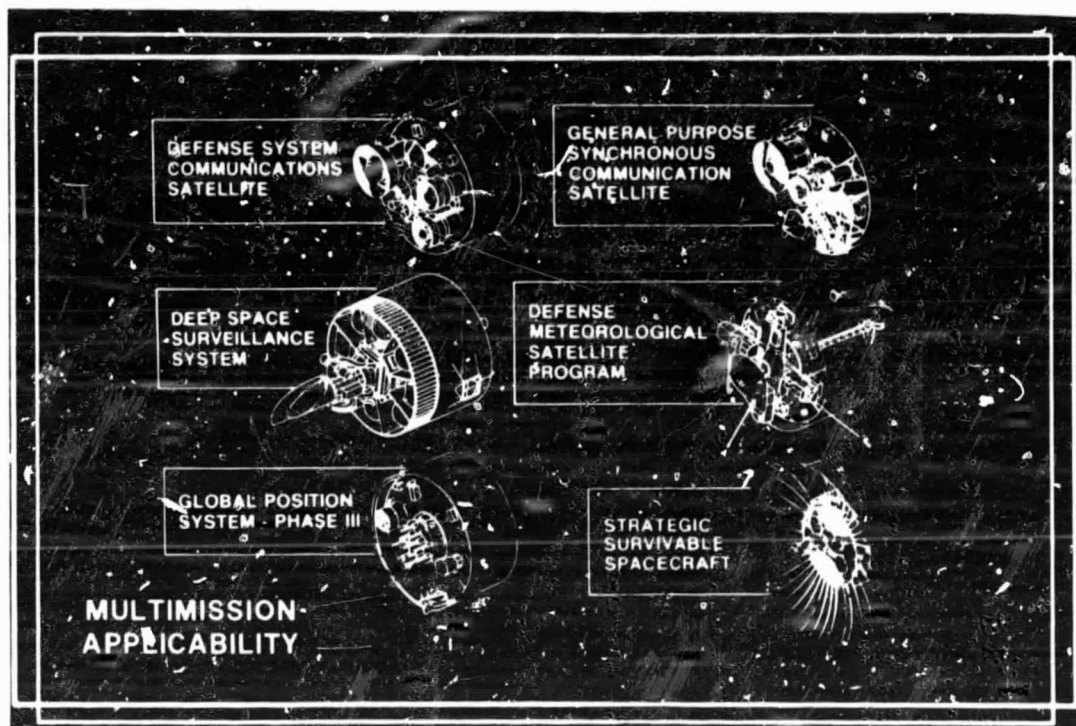
FIGURE 9   The Dynamic Isotope Power Systems (DIPS).

FIGURE 10 Flexibility of DIPS in mating with spacecraft.

FIGURE 11   SNAP reactor test experience.

MILITARY UTILITY IN SPACE
WHY NUCLEAR POWER INSTEAD OF SOLAR?

Because it enables survivability and endurability of the military spacecraft and its associated ground system, by providing:

- Survivability against nuclear attack (JCS)
  Hardness against thermal shocks from x-ray bursts

- Survivability against laser attacks (SMATH)
  Not sensitive to resultant temperature rise

- Rapid maneuverability to evade anti-satellite attacks
  Rugged structure
  No delays for fold-up

- Higher power for improved mission performance:
  - more usable payload
  - higher orbits, harder to attack
  - more ground links, harder to jam
  - harder electronics, less vulnerable
  - operation in van Allen belts without degradation
  - smaller antennas, simplified designs
  - small and mobile ground receivers for improved survivability

- Small size, improved field of view, improved pointing accuracy

Space Nuclear Power is the critical path for improved multi-service warfighting capability

FIGURE 12 Reasons for DOD interest in space nuclear systems.

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┌─────────────────────────┐
│   Secretary of Energy   │
│    Donald P. Hode!      │
└─────────────────────────┘
             │
┌─────────────────────────┐
│   Assistant Secretary   │
│    for Nuclear Energy   │
│    Shelby T. Brewer     │
└─────────────────────────┘
             │
┌─────────────────────────┐
│ Deputy Assistant Secretary │
│ for Breeder Reactor Program │
│   Gordon L. Chipman, Jr.  │
└─────────────────────────┘
             │
┌─────────────────────────────┐
│ Office of Space Reactor Projects │
├─────────────────────────────┤
│      Nuclear Technology     │
│          Materials          │
│       Quality Assurance     │
│           Safety            │
│           Budgets           │
│        Administration       │
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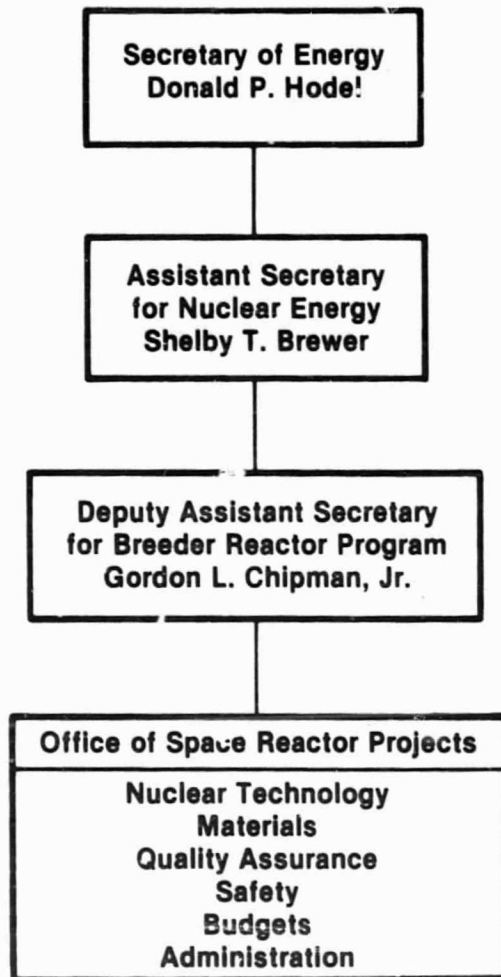
FIGURE 13   DOE organization.

- Directs all aspects of the development of advanced space reactors for use with nuclear power systems for future DOD and NASA missions.

- Develops, plans, and manages resources for the effective accomplishment of goals and objectives.

- Develops the reactor and materials technology to meet anticipated user requirements.

- Interfaces with DOD and NASA to identify missions and coordinate overall power system development.

- Directs quality assurance and quality control efforts to assure consistency with DOE and other agency requirements.

- Conducts necessary safety technology to assure adequacy of design with respect to safety and personnel exposure.

FIGURE 14   Responsibilities of the DOE Office of Space Reactor Projects.

● **Active Programs:**

   ● **HPR – Heat Pipe Reactor:**
   UO_2 fuel wafers and Mo/Re fins
   Cooled by lithium heat pipes
   Radiating to thermoelectric converters

   ● **FBR – Fixed Bed Reactor:**
   Graphite-coated UC microspheres
   Fixed annular fuel bed
   Helium coolant, Brayton cycle

   ● **RBR – Rotating Bed Reactor:**
   Same as FBR, except for fluidized fuel bed
   Rotating reactor core

● **Other Options:** Thermionic Conversion, Rankine Cycle, MHD, Gas Core

● **Power Range:**

   ● **HPR limited to lower power levels (e.g., < 500 kw(e))**

   ● **Other options best for higher power levels (megawatts)**

● **Final design selection will depend on results of ongoing and planned studies under SP-100 program.**

FIGURE 15   Space reactor alternatives.
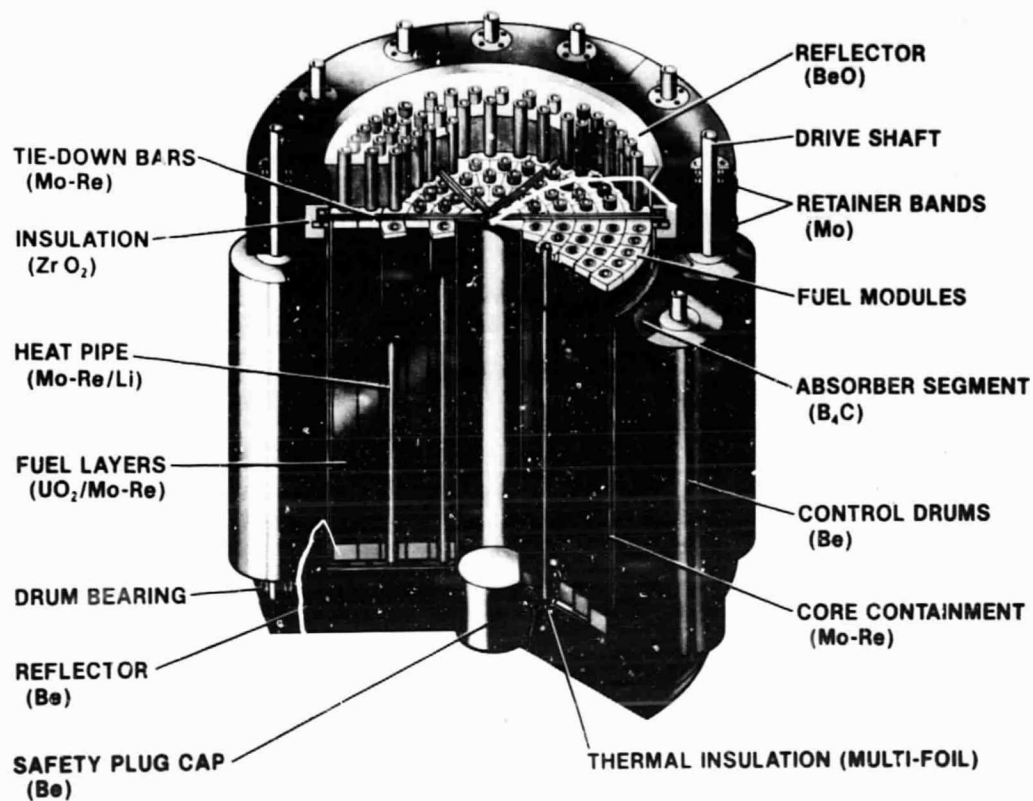
FIGURE 16   Cutaway view of Los Alamos National Laboratory HPR system.
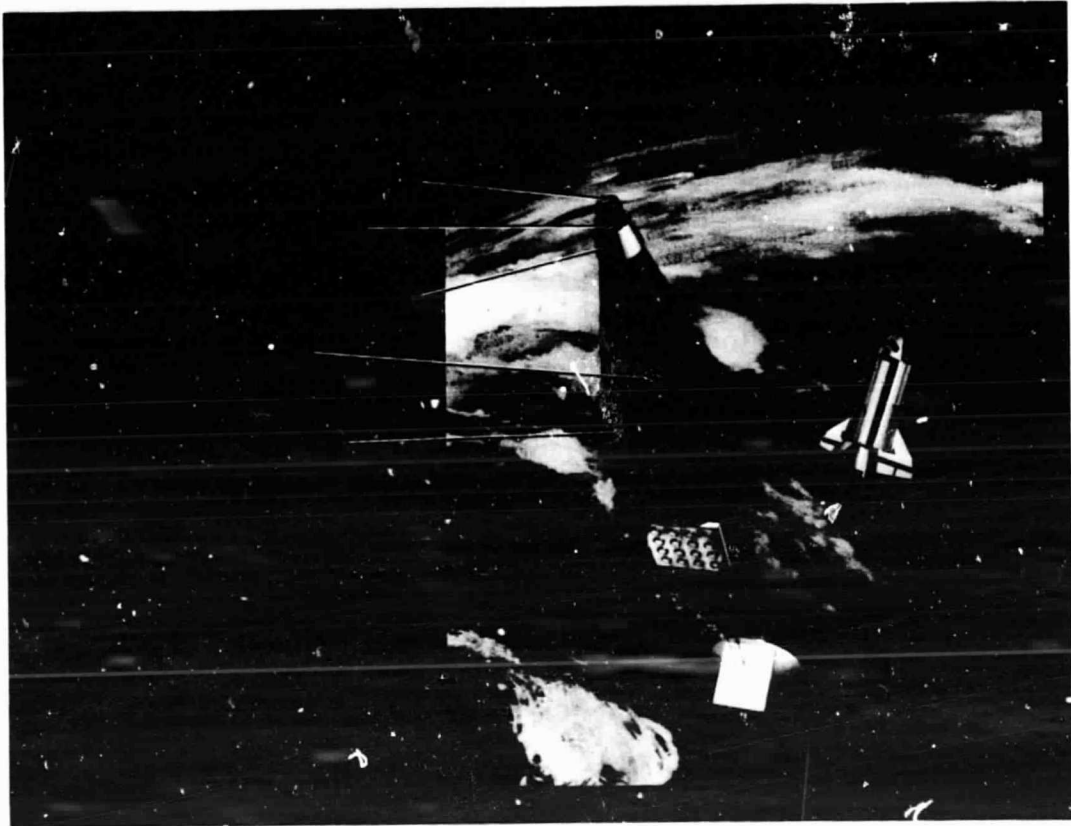
FIGURE 17   Power system design, HPR system.

- 100 kw(e) power system:
  Design scalable to lower and higher power levels

- Valuable experience base:
  Much of technology applicable to multi-megawatt systems

- Fast reactor:
  For compactness and long system life

- Heatpipe-cooled:
  For redundant heat transport

- Radiative coupling:
  Simplifies technical and programmatic approach

- Thermoelectric converters mounted on radiator:
  Simple, proven technology; High redundancy and modularity

- Lightest system at low power levels

FIGURE 18   Key features of the HPR design.
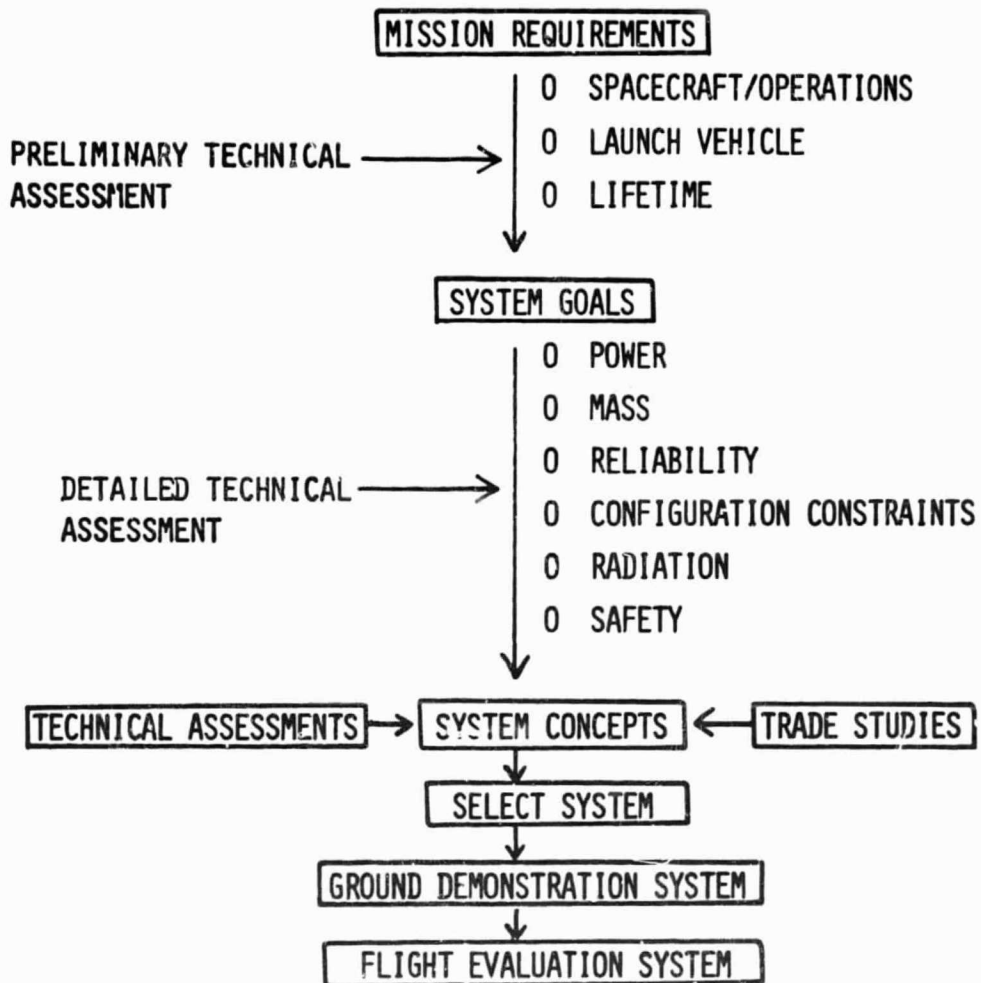
FIGURE 19 Basic steps in system planning and development.

- DOE is the US agency responsible for nuclear reactor development.

- Over the past 25 years, DOE has expended ~ $1.5 billion on space nuclear developments; about half on nuclear power and half on nuclear propulsion.

- It has developed 38 nuclear systems which were successfully used on 23 NASA and DOD space missions, and has been involved in many other space power technology programs.

- As the result of this experience, DOE personnel and support laboratories have developed a unique understanding of the pertinent technical, managerial, and flight safety issues.

- DOE looks forward to continued cooperation with DOD and NASA, to supply their nuclear reactor needs for future applications.

FIGURE 20   Summary of DOE nuclear reactor development effort.

Appendix
OUTLOOK FOR SPACE NUCLEAR POWER DEVELOPMENT

G. L. Chipman, Jr.

ABSTRACT

This paper summarizes the work conducted over the past 20 years on
space nuclear reactor system technologies. It reviews the current
activities of the Department of Energy's (DOE) development efforts,
its planned organizational structure for future space nuclear power
development, the broad range of technologies to be considered in
meeting future mission needs, and the future involvement of the
national laboratories and industrial complex in this important
national effort. The results of this survey show that DOE, with its
technical experts at the various department laboratories and
contractors, is prepared and capable of producing and delivering
flight-qualified space nuclear power systems, as it has successfully
done in the past, that would meet both civilian and military future
space mission goals.

HISTORY

Federal Responsibilities

The engineering developments during and following World War II gave
humanity great advances in three new areas of technology:
electronics, rocketry, and nuclear power. Studies in the late 1940s,
such as the U.S. Air Force's (USAF) Project Feedback, suggested the
merger of these areas of technology through nuclear-powered
spacecraft. The U.S. Atomic Energy Commission (AEC), a predecessor of
the Department of Energy, initiated a series of studies in 1951 to
address the feasibility of powering spacecraft with nuclear devices.
Parallel studies were conducted on the feasibility of using nuclear
power for the propulsion system. Out of this study phase came several
joint Atomic Energy Commission and National Aeronautics and Space
Administration (AEC/NASA) and AEC and Department of Defense (AEC/DOD)
programs to investigate the use of nuclear systems for power and/or
propulsion of spacecraft. Thus the U.S. government has long
recognized the need for having a nuclear power capability in space.
This governmental encouragement of nuclear research and development
was codified in the Atomic Energy Acts of 1946 and 1954, as amended,
which assigned to the AEC the responsibility to conduct, assist, and
foster a program of research and development to encourage the
widespread participation in the development and use of nuclear

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energy. In this way, Congress and the President focused the nation's
efforts in the nuclear area, defined the separation of powers, and
ensured the safe control of the special nuclear material needed for
these research and development programs. This concentration of
nuclear responsibility continued with the Energy Reorganization Act of
1974 and the Department of Energy Organization Act of 1977.

From this farsighted planning by the Congress and the
administration, DOE and its predecessor agencies have been able to
provide NASA and DOD with all of their space nuclear power supply
needs. The following sections review the history of the space nuclear
power program. leading up to the current status and the planning
process of the future.

Radioisotope Power Systems: Background and Future

Overview of the Nuclear Power Option

Since 1961, the United States has launched 23 civilian and military
space systems having all or part of their power requirements supplied
by nuclear power sources. DOE and its predecessor agencies have
successfully demonstrated the unique capability of extending support
to the space program through research and development in the nuclear
energy field and by providing all of the nuclear power sources flown
by the U.S. government or planned to be flown. Table A-1, a summary
of space nuclear power systems launched by the United States
(1961-1980), is useful in reviewing the history of all of the flight
space nuclear power systems provided by the department.

Looking to the future, it is apparent that broad ranges of power
output and lifetime will be required to comply with the developing
range of mission needs. The functional missions flown to date have
required low power and long life; these needs have been met by using
radioisotope thermoelectric generators (RTGs). As future power needs
rise to 1 kW(e) and above, the higher conversion efficiency of dynamic
electric conversion systems will become increasingly attractive,
making them the preferred power system. For example, conversion
efficiencies of 15-25 percent are possible with dynamic systems
compared to 5-10 percent for thermoelectric systems. A ground
prototype system of the Dynamic Isotope Power Systems (DIPS) has been
built and operated for 5,000 hours. At a power level of about 10
kW(e), the increasing inventory of fuel needed in the DIPS will make
reactor systems the preferred system. At 25 kW(e) and above, reactor
systems become more attractive in cost, mass, and size than all other
competing power systems. Figure A-1 illustrates where the transition
power levels occur and how the three classes of nuclear power systems
are related to development time and mission needs.

TABLE A-1 Summary of Space Nuclear Power Systems Launched by the United States (1961-1980)

Power Source[a]	Spacecraft	Mission Type	Launch Date	Status
SNAP-3A	Transit 4A	Navigational	June 29, 1961	Successfully achieved orbit
SNAP-3A	Transit 4B	Navigational	Nov. 15, 1961	Successfully achieved orbit
SNAP-9A	Transit-5BN-1	Navigational	Sept. 28, 1963	Successfully achieved orbit
SNAP-9A	Transit-5BN-2	Navigational	Dec. 5, 1963	Successfully achieved orbit
SNAP-9A	Transit-5BN-3	Navigational	April 21, 1964	Mission aborted; burned up on reentry
SNAP-10A (reactor)	SNAPSHOT	Experimental	April 3, 1965	Successfully achieved orbit
SNAP-19B2	Nimbus III	Meteorological	April 14, 1969	Successfully achieved orbit
SNAP-27	Apollo 12	Lunar	Nov. 14, 1969	Successfully placed on lunar surface
SNAP-27	Apollo 13	Lunar	April 11, 1970	Mission aborted or way to moon. Heat source returned to South Pacific Ocean
SNAP-27	Apollo 14	Lunar	Jan. 31, 1971	Successfully placed on lunar surface
SNAP-27	Apollo 15	Lunar	July 26, 1971	Successfully placed on lunar surface
SNAP-19	Pioneer 10	Planetary	March 2, 1972	Successfully operated to Jupiter and beyond
SNAP-27	Apollo 16	Lunar	April 16, 1972	Successfully placed on lunar surface
Transit-RTG	Transit (TRIAD-01-1X)	Navigational	Sept. 2, 1972	Successfully achieved orbit
SNAP-27	Apollo 17	Lunar	Dec. 7, 1972	Successfully placed on lunar surface
SNAP-19	Pioneer 11	Planetary	April 5, 1973	Successfully operated to Jupiter and Saturn and beyond
SNAP-19	Viking 1	Mars	Aug. 20, 1975	Successfuly landed on Mars
SNAP-19	Viking 2	Mars	Sept. 9, 1975	Successfully landed on Mars
MHW	LES 8	Communications	March 14, 1976	Successfully achieved orbit
MHW	LES 9	Communications	March 14, 1976	Successfully achieved orbit
MHW	Voyager 2	Planetary	Aug. 20, 1977	Successfully operated to Jupiter and Saturn and beyond
MHW	Voyager 1	Planetary	Sept. 5, 1977	Successfully operated to Jupiter and Saturn and beyond

[a]SNAP stands for Systems for Nuclear Auxiliary Power. All odd-numbered SNAP power plants use radioisotope fuel. Even-numbered SNAP power plants have nuclear fission reactors as a source of heat. MHW stands for the Multihundred Watt RTG. LES stands for Lincoln Experimental Satellite.

SOURCE: G. Bennett, Overview of the U.S. Flight Safety Process for Space Nuclear Power, Nucl. Safety, 22(4), 424, 1981.

Radioisotope Thermoelectric Generators

The RTG power system has been a cornerstone of the space nuclear program. Currently, DOE has under development the general-purpose heat source RTG that is planned for use on the NASA Galileo and International Solar Polar missions scheduled for launch aboard the Space Shuttle in 1986. Also being developed is the technology for the modular isotope thermoelectric generator (MITG), potentially to be used in U.S. space missions beginning in the 1988 time frame. An estimated $45 million and 4 years would be required to complete the research and technology verification phase of this advanced RTG system. Table A-2 identifies the progress made in the isotope thermoelectric power systems and the objectives DOE has set for future development.

Dynamic Isotope Power Systems

To meet potential DOD future space missions forecast in the 1- to 2-kW(e) power range, DOE embarked on a development program to evaluate so-called dynamic conversion systems that employ a rotating turbine/alternator system that offers efficiencies approaching 18-25 percent. Following a competitive phase wherein studies and tests of the Brayton cycle and the Rankine cycle were supported, DIPS, based on an isotope-fueled closed Rankine power conversion cycle utilizing the organic compound Dowtherm A as the working fluid, was selected (see Figure A-2). To date, prototypic equipment has been operated in excess of 5,000 hours in a space vacuum. Approximately $20 million was spent during the course of this program. Figure A-3 illustrates the flexibility of this type of space isotope power system in mating with the Nuclear Integrated Multimission Spacecraft (NIMS). NIMS can accommodate the four major identified DOD mission categories, i.e., communications, surveillance, navigation, and meteorology. An estimated $40 million and about 3 years would be required to qualify this system for space flight.

The Space Reactor Program Through 1973

For the space nuclear reactor power system development efforts, the AEC expended some $1.4-$1.5 billion on the following nine areas through 1973, when all activities associated with space nuclear reactor power development ceased.

SNAP-2: The development of this zirconium hydride reactor coupled to a 3-kW(e) mercury Rankine cycle combined rotating unit (CRU) was initiated in fiscal year (FY) 1956 at the request of the Air Force. Two reactor tests demonstrated reactor feasibility. Congressional action on the DOE FY 1964 budget necessitated the elimination of the proposed SNAP-2 launch program. The AEC terminated the SNAP-2 program

TABLE A-2   Trends in RTG Technology

Generator[a]	SNAP-3	SNAP-27	SNAP-19	MHW	GPHS-RTG	MITG
Mission	Transit 4	Apollo	Pioneer, Viking	LES 8/9, Voyager	Galileo, Solar Polar	Future
Power (W)	2.5	65	40	150	285	300
Fuel form	Pu metal	PuO_2 micro-spheres	PuO_2-Mo cermet	Dense	Dense	Dense
Fuel clad	Ta	Super-alloy	T-111	Ir	Ir	Ir
Thermo-electrics	PbTe	PbSnTe	PbTe-TAGS	Si-Ge	Si-Ge	Si-Ge or Si-Ge/GaP
Conversion efficiency	5.0	5.0	6.3	6.7	6.8	8.2
Specific power (W/kg)	1.48	2.34	3.00	3.94	5.14	9.05

[a]MHW, multihundred watt; GPHS-RTG, general-purpose heat source radioisotope thermoelectric generator; MITG, modular isotope thermoelectric generator.

in FY 1964 but continued the development of the zirconium hydride reactor technology and the mercury Rankine cycle technology. The mercury Rankine effort was terminated in FY 1967.

SNAP-10A: The development of the SNAP-2 zirconium hydride reactor coupled to a direct radiating thermoelectric conversion system was initiated in FY 1960 at the request of the Air Force. Power output was 500 W(e). Congressional action on the DOD FY 1964 budget necessitated the elimination of the proposed SNAP-10A launch program. Because of the importance of the SNAP-10A flight test to the nation's nuclear electric power effort, the Joint Committee on Atomic Energy (JCAE) included the authorization for the flight test by the AEC in the FY 1965 Authorization Act, and funding was appropriated. The SNAP-10A system was launched on April 3, 1965, and operated successfully for 43 days prior to shutdown, caused by a spurious signal from the payload. The program was concluded in FY 1966 except for the completion of the 10,000-hour ground test of a duplicate of the flight system as part of the zirconium hydride reactor technology program. The key fact to be remembered is that SNAP-10A was the first known use of a nuclear reactor in space--and it was successful. The SNAP-10A experimental flight proved that space nuclear reactors can be safely built, launched, and operated remotely.

Liquid-metal-cooled reactor (SNAP-50-SPUR): This technology development was initiated in 1958 under the Aircraft Nuclear Propulsion (ANP) program in an attempt to meet the design requirements for supersonic flight. The concept was a high-temperature, lithium-cooled refractory metal alloy reactor. The ANP program was terminated in FY 1961, but a technology effort was continued because of DOD and NASA interest in the development of a high-power nuclear electric system for space application. In FY 1962, the program was directed toward the Air Force SPUR concept that contemplated a 2000°F, fast, lithium-cooled refractory alloy reactor coupled to a potassium Rankine cycle power conversion system. A SNAP-50/SPUR office was established in AEC headquarters with an Air Force officer as program manager and with NASA, Air Force, and AEC deputy managers. Because the time could not be defined when high power in space applications would be required, the character of the program changed during FY 1964-1965 from a power development approach to a long-term basic technology program, leading to a ground demonstration of the reactor and critical components of the power conversion system. The SNAP-50/SPUR program was terminated in FY 1965, and the responsibility for the technology program was transferred from Pratt and Whitney (CANEL) to Livermore (LRL). In FY 1968, owing to a reduction in funding because of budgetary pressures, the program was terminated, as directed by the Joint Committee on Atomic Energy.

Rover/NERVA: Perhaps the most challenging application of nuclear energy to spacecraft was the nuclear rocket program. G. T. Seaborg summed up the challenge as follows: "What we are attempting to make is a flyable compact reactor, not much bigger than an office desk,

that will produce the power of Hoover Dam from a cold start in a matter of minutes."

The United States carried out an ambitious research program from 1955 to 1972 aimed at developing a capability to use nuclear power for rocket propulsion. This program, which was jointly managed by AEC/USAF and AEC/NASA, demonstrated a number of important firsts in space nuclear technology: (1) multiple-start capability, (2) peak power of 4,200 MW, (3) 60-min continuous operation, sufficient for many space missions, and (4) ability to start on its own power and operate stably over a wide range of conditions. From this research evolved the nuclear engine for rocket vehicle applications (NERVA) that was designed for use on a reusable nuclear shuttle. NERVA was to produce 75,000 lb of thrust at a specific impulse of 825 s--twice the specific impulse of the best chemical rockets. Such a capability trannslates into larger payloads and shorter flight times. Furthermore, NERVA offered the possibility of producing 15-25 kW of power for hotel functions during the coast phase of the mission.

The Rover/NERVA program demonstrated again that several government agencies can cooperatively work together to apply nuclear technology to spacecraft and that nuclear power can provide a quantum leap in performance improvement over more conventional systems.

Boiling metal reactor (MPRE): This technology involved the investigation of the feasibility of direct boiling of potassium in a compact fast-spectrum reactor. The decision was made in FY 1966 to phase out this work and concentrate the remaining advanced technology resources on the gas-cooled, liquid-metal-cooled, and thermionic concepts. The program was terminated at Oak Ridge National Laboratory (ORNL) in FY 1966.

Gas-cooled reactor (710): This technology involved the development of a high-temperature, lightweight, high-performance, gas-cooled reactor suitable for a Brayton-cycle space power system. During FY 1967 the work was reduced to a fuel element development program only. In FY 1968, the program was terminated at General Electric in order to apply available resources to the liquid-metal-cooled and thermionic reactor concepts.

SNAP-8: The SNAP-8 program was initiated in FY 1960 to develop a 30- to 60-kW electric system suitable for space propulsion. The AEC sponsored development of the zirconium hydride reactor heat source, and NASA was responsible for the mercury Rankine power conversion system. The S8ER (SNAP 8 Experimental Reactor) operated for 1 year during FY 1964-1965, and the S8DR (SNAP 8 Developmental Reactor) operated for 7,000 hours during FY 1969-1970.

The 5-kW(e) zirconium hydride/thermoelectric system: The objectives of the ZrH/TE work were to fabricate and test a "5 year life" reactor and power conversion system. Nominal power output was to be 5 kW(e). The NASA-Lewis Research Center was responsible for program management, and the AEC funded reactor development at Atomics International. The program was terminated in January 1973.

Thermionic reactor: The thermionic reactor was a concept for converting heat to electricity inside a reactor core using the thermionic process. It can cover the range from a few kilowatts to several megawatts of electric power, but the emphasis in the last several years of activity was on a 120-kW system suitable for space propulsion missions.

The AEC program began in FY 1959 with support of the Los Alamos Scientific Laboratory (LASL) "plasma thermocouple." In FY 1964, the AEC reoriented the thermionic program to emphasize industry participation (General Electric and General Atomic) in reactor fuel element development and eventual construction of a reactor experiment. In August 1970, Gulf General Atomic was selected for the prime role of developing the TFE (thermionic fuel element) and constructing a thermionic reactor test. The AEC and NASA thermionic efforts were administratively combined in FY 1971, with NASA assuming the responsibility for systems studies and support technology. All space thermionic effort by AEC was terminated in January 1973.

Dynamic power conversion: This technology was directed toward three Rankine cycle concepts--mercury, potassium, and organic. The mercury Rankine effort was originally started as part of the development of a nuclear reactor (SNAP-2) for the Air Force. It was separated from the reactor development effort and conducted as a separate technology from FY 1964 through FY 1973, at which time it was terminated. The potassium Rankine effort was funded by the Air Force through FY 1963, became an AEC responsibility as part of the SNAP-50/SPUR concept in FY 1964 and was terminated in FY 1968. Finally, the organic Rankine effort began in FY 1966 in response to a DOD interest and continued through FY 1970, when it was terminated by the AEC with the understanding that the Air Force would fund any future effort in this area.

Reactivation of Space Reactor Development Efforts

In 1975, a DOD/ERDA Space Nuclear Application Steering Group was organized with the objective of establishing and maintaining the necessary management interface and communication channels between DOD and the Energy Research and Development Administration (ERDA) for the purpose of ensuring the effective and efficient military use of nuclear energy for space and other directly related applications. In concert with the steering group, an Advanced Space Power Working Group was established and provided a report, SPWG 77-1, in 1977. This report assessed DOD potential missions and the various space power technologies and formed the basis for DOD to enter into a 5-year space reactor technology program beginning in 1979 at approximately $2 million/yr. At that time, DOD space system power requirements appeared in the 10- to 100-kW(e) range. The technology development program focused on a reactor system concept called Space Power Advanced Reactor (SPAR).

In 1981, during the formulation of the FY 1982 budget submission, the Office of Management and Budget (OMB) directed that DOE funds for space reactor development be held to $1 million and be contingent upon support from DOD and/or NASA. While DOD decided to continue to study its mission applications and could offer no financial support to DOE, NASA pledged to support the program and to venture toward a joint NASA/DOE technology verification phase of the reactor development project, redesignated SP-100. The NASA mission models indicated that 100 kW(e) could provide a suitable power level for both outer planetary and earth orbital missions. In view of the budgetary constraints on DOE, the work associated with the power conversion subsystem was planned to be conducted and supported by NASA, while the efforts on the reactor subsystem would continue to be conducted by DOE based upon DOD and NASA support. In FY 1982, with NASA support, the reactor development program was tailored to NASA objectives that identified a number of mission considerations for the 100-kW(e) power system.

CURRENT STATUS OF SPACE REACTORS

SP-100 Driven by NASA

As was stated earlier, the origins for initiating the technology of an advanced space power reactor development program were based upon the potential DOD missions in the 10- to 100-kW(e) power range. Firm requirements for power above 100 kW(e) have not been officially identified to DOE by DOD. With the advent of active NASA involvement in the program, the characteristics of the development program became oriented toward the civilian missions. For example, need for a 7-year, continuous operation at full power followed by a 5-year reduced power output became important. Also, the attributes of a thermoelectric conversion system that would provide high reliability and redundancy in avoiding single-point failures appeared very desirable. In addition, the desire to provide 100 kW(e) from a single shuttle launch of an integrated spacecraft dictated a specific power minimum of 36 W/kg and a goal of 55 W/kg. Naturally, these power system attributes might be useful to such potential DOD missions as space-based radar, surveillance, communications, electric propulsion, and jammers. On the other hand, potential DOD missions such as lasers, particle beams, and advanced concepts in the 1-MW to hundreds-of-megawatt pulsed power level probably could not be satisfied by the current heat pipe/thermoelectric type of power system.

DOD Interest (Increasing)

Kilowatt Range

The early mission analysis effort completed in 1977 indicated the potential need for an advanced space reactor power system in the 10- to 100-kW(e) power range. Currently, it appears that these defense needs could span the power range from 5 to 400 kW(e). Over the past 5 years the forecast power levels required in space have increased substantially, and with the added emphasis on survivability, the need for space reactor power becomes more evident. It would be well, however, to raise a note of caution that until detailed mission analyses and trade-off studies are accomplished, these potential projected power levels could vary substantially within the power envelope.

Megawatt Range

The use of lasers, particle beams, and other advanced energy-dissipating systems might require pulse power in the 1- to 100-MW range. Space nuclear reactor power systems that might have the capability of providing such high pulsed power may exist in the technology that was supported via the nuclear rocket technology program. Such reactors coupled to an open-cycle Rankine-alternator conversion system or the rotating-bed reactor and/or gaseous-core reactor concepts when coupled with closed- or open-cycle electric energy conversion devices should be considered. Each of these types of technologies has its advantages and disadvantages. It is clear that a space nuclear capability will require the consideration of many technical alternatives, and more feedback on the mission plans and power needs.

DOE Organization

In view of the importance that DOE places on the space reactor program, the Assistant Secretary for Nuclear Energy plans to establish an Office of Space Reactor Projects, reporting to the Office of the Deputy Assistant Secretary for Breeder Reactor Programs. The new office will be responsible for the management and coordination of all activities in DOE's space reactor development program. The office will consist of a cadre of technical personnel who will be assisted by matrix support from the department's nuclear sector in the areas of materials performance, quality assurance, safety, budget, and administration.

The new office will direct DOE's effort to develop the space reactor technology required to meet user agency requirements. The office will work closely with the user agencies, DOD and NASA, to

ensure proper integration of reactor development activities with mission requirements and power conversion system development functions.

The near-term efforts of the office will focus on a review of the past and present space reactor development activities and technologies. A variety of technologies will be examined to ensure that the reactor development efforts to follow will emphasize those concepts with the highest possible likelihood of successful mission performance. This review effort will then form the basis of a space reactor technology plan that will be coordinated and agreed upon with the user agencies and will be updated on a regular basis. The current emphasis on space reactor development reflects an increasing awareness of the need for higher power levels in future NASA and DOD missions.

Reevaluation of Space Reactor Technology

As was noted in the preceding section, one of the early activities of the Office of Space Reactor Projects will be to reevaluate past and present space reactor activities for relevance to future needs. This review will focus on those technologies that offer a contribution to future high-technology systems. For example, the ZrH reactor systems are limited in temperature and power density capabilities and do not appear to offer much in reactor core technology for the future. Many reactor systems appear to show more promise.

Reactor Technologies of Interest

RBR/FBR: The Rotating-Bed Reactor (RBR) and the Fixed-Bed Reactor (FBR), which evolved from the rotating-bed nuclear rocket concept, are closely related systems. Figure A-4 is a schematic of the reactor in its rocket format. In both power systems, the reactor core consists of an annular cylindrical bed of unbonded fuel particles. The coolant/working fluid is a gas that passes radially inward through the fuel bed and exits axially from a central cavity or annulus. Preferred gases are hydrogen and helium. The inherent advantages of this type of system lie in the use of particulate fuel. By making the particles small, very high heat transfer can be obtained; by keeping them unbonded, the core becomes highly resistant to thermal shock and can withstand high-rate thermal transients. Present concepts utilize U-ZrC particles in the 200- to 600-μm-diameter range.

In the RBR format, the original rocket concept, the particle bed is retained in its annular cylindrical shape by centrifugal force; it is held against an external porous cylinder by being rotated about its central axis. The bed can be fluidized to any desired degree by control of the cooling gas velocity and rotational speed. The FBR format is simpler; the particle bed is retained in a close-packed array between inner and outer porous cylinder, and the core is not rotated. It is apparent that in the FBR format, maximum gas

temperature is limited by the requirement that the inner porous cylinder survive. When the reactor is to be used to drive a turbine, this limitation on temperature is not significant, and the simplicity of the FBR format becomes attractive. When maximum temperatures are utilizable, as, for example, in an open-cycle magnetohydrodynamic (MHD) system or in rocket propulsion, the additional complexity of the RBR format may become worthwhile.

The RBR/FBR reactor concept is very flexible and can probably be adapted to many different types of power systems. It can be coupled with closed-cycle Brayton conversion systems for continuous power over a wide range of power levels; it can be coupled with open-cycle Brayton or MHD conversion systems for pulsed power outputs; one can even postulate a dual-purpose system in which the high-temperature gas from the reactor could be used alternatively for electrical power or for evasive maneuvering by being discharged through a rocket nozzle.

The reactor concept merits careful evaluation, and a study program is being carried out by DOE's Brookhaven National Laboratory under Defense Advanced Research Projects Agency (DARPA) funding.

HTGR: The High-Temperature Gas-Cooled Reactor (HTGR) is a large stationary power plant system, but the materials technology, much of it developed on the program, is very relevant to the space reactor effort. HGTR is a helium-cooled system with much of the core fabricated of graphite. A large part of the original work on the development of high-density graphites, on the development of fabrication methods, and on the measurement of physical properties was done on the program. Thermal properties of graphite and rates of permeation and diffusion of fuel and fission products were determined that can be utilized in many space reactor concepts. The HGTR fuel consisted of U-ZrC fuel particles clad with pyrolytic carbon, and such particles are prime candidate for RBR/FBR fuel particles. Also, graphite is likely to be a favored structural material for many components in some space reactor concepts because of its unique high-temperature properties. The graphite and fuel particle technology developed on HTGR is a valuable resource for a continuing space reactor program and should be reviewed and reevaluated thoroughly.

KIWI/Rover/NERVA: This program was the major AEC effort to develop a nuclear rocket engine that was active from 1955 to 1972. During this period, approximately 18 distinct reactor systems were constructed and operated, culminating in NRX-A6, which operated at an average power level of 1,155 MW for over 1 hour. The Phoebus-2A demonstrated the capability of operating at 4,200 MW. These tests showed that the nuclear rocket concept has the capability of reaching high power levels quickly and of sustaining them for significant times. The total expenditure of funds in this effort was approximately $1.4 billion, through AEC, NASA, and DOD funding.

The reactors built and operated differed a great deal in size and design details, but all were solid-core systems constructed primarily of graphite and cooled by gaseous hydrogen. The fuel was particulate

UC$_2$, clad with and and embedded in graphite. As in the HTGR program, a great deal of graphite technology was generated that constitutes a resource for future high-temperature reactor developments.

In addition, a wealth of experience was gained with the operation and control of reactor systems with very high rate transients of temperature and power that should also be most valuable for pulsed power space systems. Furthermore, the hydrogen handling technology developed for use under these extreme conditions should be valuable.

Liquid-metal systems: The most familiar utilization of liquid metals in reactor systems has been as a heat transfer medium, the primary advantage being that liquid metals can move large quantities of heat at high temperatures with low vapor pressures, leading to inherently light, high-power-density systems, obviously attractive for space reactors. The potential difficulties that must be overcome to develop a successful liquid-metal-cooled system are the tendency of the liquid metals to dissolve their plumbing, a high susceptibility to very small levels of contaminants, and the general difficulty of keeping auxiliary devices like valves, pumps, bearings, and instrumentation working reliably.

A substantial number of ground-based reactors have been designed, built, and operated that can provide many background data on the techniques of operating liquid-metal systems. These include EBR-I, EBR-II, FFTF, Enrico Fermi, and Sea Wolf. In addition, a number of advanced technology programs have contributed to the field, including the Pratt and Whitney effort in the ANP/SNAP-50/SPUR program and the Molten Potassium Reactor Experiment (MPRE) at ORNL. All of these programs should be reviewed for their contributions to the data base.

Fluid-fuel systems: For advanced pulse power systems, one should also consider reactor systems where the fuel itself is liquid, or even partially gaseous. Although establishing a standby mode may be difficult, such systems could also provide high rates of heat transfer and great resistance to thermal shock. Liquid-fuel systems have been studies in the LAMPRE and Molten Salt Reactor Experiment (MSRE) programs, and a partially gaseous core concept was studied under the Plasma Core Nuclear Rocket effort. These programs should be reviewed for their technological content.

Power System Information and Data Correlation Study

As a start in evaluating reactor technology, DOE has contracted with the Energy Systems Group of Rockwell International to perform a power technology assessment, mission assessment, and program planning study for space nuclear reactor power systems. There will be three phases of power technology analysis: information collection, data correlation, and power system synthesis. In carrying out this analysis, each power system will be regarded as consisting of five subsystems and will be analyzed in those terms. The five major

subsystems are reactor, shielding, primary heat transport, power conversion, and heat rejection. An important aspect of information collection and data correlation will be the determination of the level of technology readiness that exists for each subsystem of each power system, and a corresponding estimate of the development time and funding required to bring the subsystem to flight readiness.

Once the power technology assessment is complete, Rockwell personnel plan to develop an understanding of mission planning and needs by a similar three-phase process, namely:

o <u>Mission information collection</u>: This will include a search of the literature and contacts with the users.

o <u>Mission data validation</u>: This will include further work with users to validate a set of missions requiring space nuclear reactor systems as well as launch and spacecraft interface/integration.

o <u>Mission scenario synthesis</u>: This process will use the validated mission data to develop three or more mission scenarios categorized into the near term, midterm, and far term.

The final stage of the study (power system synthesis) will consider technology readiness relative to mission needs. Three power systems will be conceptually defined: early deployment (late 1980s), midterm deployment (early 1990s), and late deployment (beyond 1955).

This is the first step in the larger process of establishing the mission power requirements. But when one considers the large lead times to develop new technology, this is a very critical and timely step. The results of this study should provide a valuable baseline for further planning efforts.

Continuing SP-100 Effort

As noted in the section on reactivation of space reactor development efforts, the SP-100 program (originally called Space Power Advanced Reactor, or SPAR) was initiated in 1979 in response to DOD interest in power levels of 10-100 kW(e) and reinforced in 1981 by NASA interest in power levels of 100 kW(e). The design has a nominal output of 100 kW(e) but is scalable up or down. It is a fast reactor fueled with fully enriched UO_2, utilizing heat pipes for heat transport and thermoelectrics for power conversion. A moderate amount of reactor design work has been done, including a successful criticality experiment. In 1981, DOE and NASA agreed to work cooperatively on the system, DOE concentrating on the reactor, shield, and heat transport system and NASA assuming responsibility for the thermoelectric converter subsystem, including the radiator.

The core evolved from the use of heat pipes for heat removal and has, as a result, a novel "inside out" fuel element configuration. Figure A-5 shows a heat pipe fuel element "module," the equivalent of a fuel rod in a conventional core. The core consists of 120 of these

modules in close-packed cylindrical array; the modules closely
approach each other (some clearance is left to accommodate fuel
swelling) without any separating clad or coolant. Heat is thus
extracted centrally from each "fuel rod" rather than from the external
surface, as is more usual. The heat pipes are constructed of Mo-13Re,
as are the attached fins, which serve to reduce the maximum
temperature in the UO_2. The working fluid is lithium. The heat
pipes extend out of the core, through or around a shadow shield, and
radiate to the thermoelectric conversion system. Figure A-6 shows how
the heat pipe fuel element modules are assembled into the reactor;
Figure A-7, how the reactor, shield, and converter/radiator are
assembled into a power system.

Present DOE activities are focused primarily on the experimental
development of the heat pipe design details and fabrication methods,
and on an in-pile irradiation experiment that simulates the heat pipe
fuel element module. This will explore fuel swelling in this novel
configuration and also the behavior of the heat pipe materials in an
in-core environment. The program has been operating at $2 million/yr;
continuation at only a modestly increased level for the next 3 years
is contemplated. It thus amounts to a technology development effort
rather than a serious attempt to construct a reactor system in the
immediate future. The planning process to be described in the next
section will indicate which reactor power system concept should be
pushed to flight readiness. This will require substantially higher
funding than the current rate of expenditure.

PLANNING PROCESS

Having described the history of space nuclear power and its current
status, it is worthwhile considering where we go from here. It is
important to establish in the beginning a rational planning process so
that future space nuclear power systems will meet future
requirements. Figure A-8, which is an outgrowth of good systems
engineering practice, is a schematic of the basic process one should
follow in developing a flight-qualified space nuclear power system.
These are, to quote Wilton P. Chase, "...the irreducible gross
functional steps which must be followed."

Mission Assessment

The key to good design is to establish the mission requirements at the
start of the design work. This statement sounds so obvious as to be
dismissed by most designers, yet it is amazing how often the essential
nature of the mission requirements is forgotten. Once the
requirements are established, the designer is in a much better
position to select the system concept to satisfy these requirements.
As was noted in the section on the current status of space reactors, a

number of studies have shown the need for larger power systems to supply power to future space missions. These can be categorized as follows:

Orbital applications:

o   Communications systems requiring only small, low-power, earth-based transmitters/receivers
o   Remote sensing of the earth (e.g., improved Landsats)
o   Electrical power supply for large, permanent, manned space bases
o   Active defense systems.

Space exploration:

o   Nuclear electrical propulsion
o   Electrical power supply for manned or unmanned deep-space probes
o   Electrical power supply for bases established on planetary bodies (lunar surface, Mars, etc.).

What is needed now, if we are to have the power technology available for application when it is needed, is to review the missions that are being planned and determine their power requirements. An improved understanding of mission needs is required in order to expend further space reactor funds effectively.

Power System Technology Assessment

As was indicated in the section on reevaluation of space reactor technology, a wide variety of power system concepts are possible, potentially employing a broad spectrum of technologies. The various potential subsystems exist at widely differing degrees of readiness. The cost in time and dollars to develop a flight-ready power system will vary from moderate to significant, depending on system complexity. For an order of magnitude guess, one might estimate 10 years and $300 million for a moderately complex system, and 15-20 years and $1,000 million for a complex advanced system. It would be imprudent to embark on any such development effort without a firm understanding of mission requirements.

It can be argued that, in the absence of well-defined mission requirements, efforts ought to be concentrated on some of the more promising technology areas. This is, in effect, being done at present, with work going on directed toward heat pipe reactors and the evaluation of particulate bed reactors. But such efforts are inherently limited, first by the process of getting budgetary support for nonmission technology and second by natural caution and differences of opinion among planners.

Some areas of technology suggest substantial benefits in specialized applications but also suggest high costs, which mitigate

starting any effort in advance of mission definition. One such area
is the droplet radiator. As power system output rises beyond a few
megawatts, the weight and size of a conventional radiator dominate the
entire system, indicating that perhaps a radical new approach to the
problem of dumping waste heat is required. The droplet radiator may
be such a new approach. But it promises to be a difficult, expensive,
and lengthy development item. Without better advanced planning, this
subsystem could well become the limiting component of a required power
system in the next decade.

It is true that at low power levels some time can be bought at the
price of performance. Heavier, shorter-lived power systems can be
developed to flight readiness by deferring obvious technology
improvements to a "Mark II version" of the system. But the ability to
follow this route is quite limited. In most attractive system
concepts, time and money are not commutative. Much technology remains
to be developed, and much of the effort will be directed to system
design. Sound engineering practice demands an understanding of system
goals in advance of system development.

Selection of Candidate Power Systems

Referring to Figure 8, one can see that the procedure preferred by
DOE--and, further, believed by DOE to be the only sound and prudent
procedure to select a space nuclear power system for development at a
substantial cost--is (1) to understand the requirement, (2) to
synthesize several concepts able to meet the need, and (3) to select
one system for development after a careful weighing of relative costs
and benefits between the several concepts, all of which were deemed
capable of meeting the requirements. Following this procedure, DOE
therefore solicits an expansion of mission definition efforts by
potential users.

Development of Reactor Technology and Reactors

Given the selection of the preferred power system, as described in the
preceding section, it becomes the function of the DOE Office of Space
Reactor Projects to proceed in an orderly fashion with the development
of the reactor subsystem to the space nuclear power system. An early
step in this process will be the generation of an "interface
document," clearly defining the functional requirements of the reactor
subsystem and describing quantitatively how it intergrates with the
rest of the power system. Typical of this process is the SP-100 power
system, in which DOE is responsible for the reactor, the shield, and
the heat transport to the conversion system, while NASA is responsible
for the conversion system and radiator.

With the limit of DOE responsibility adequately defined, it is
possible to generate an integrated plan to cover the sequential

development of all key components, including such technology development and verification as is required to support the reactor system development. The reactor design will be continuously refined and updated as the component performances are better established.

DOE READINESS

DOE is charged by law with responsibility of all reactor development programs. DOE has had years of experience with the development of many specialized reactor systems, prototypical of space nuclear power systems. This work has been accomplished through the efforts of many people--managers, engineers, craftsmen, technicians--in the national laboratories and in private industry. National laboratories that have been particularly active in the field of space nuclear power include Los Alamos, Oak Ridge, Savannah River, Brookhaven, Livermore, Argonne, Sandia, Hanford, Idaho National Engineering Laboratory, and the Mound Facility. Industrial participants range from very large to quite small and include General Electric, Westinghouse, Rockwell International, Aerojet, General Atomic, TRW, Thermo Electron, Teledyne Energy Systems, Fairchild Industries, and many others. These widespread activities have created over the years a large and diversified population of technical experts with interest and experience in advanced nuclear power systems. This resource is waiting to be reengaged and redirected in a new effort to put nuclear reactor power to work in space.
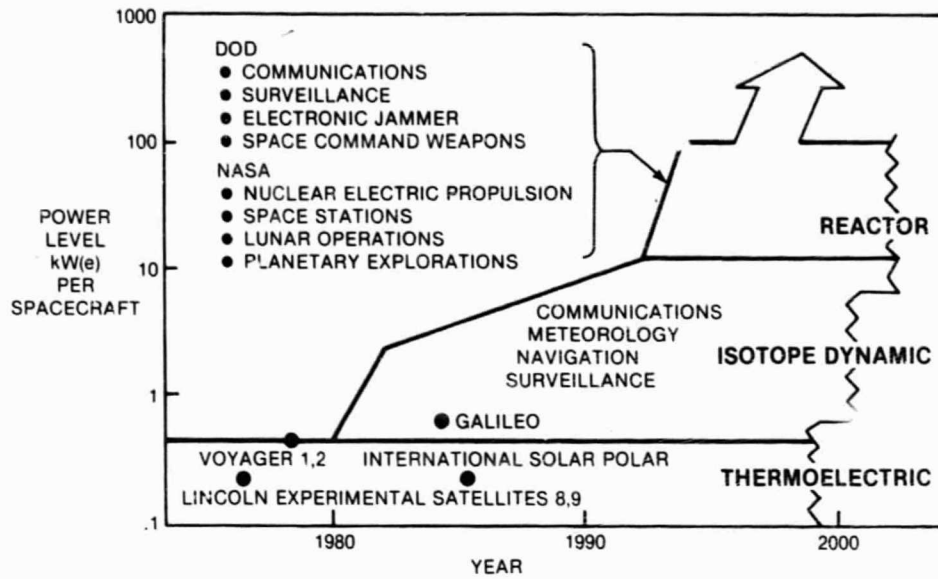
FIGURE 1 Space nuclear power and evolving mission requirements.
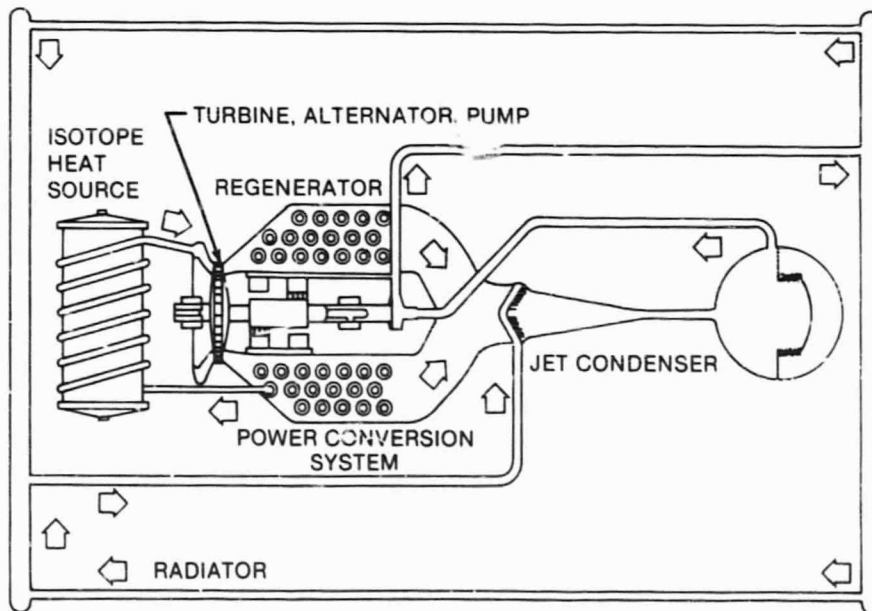

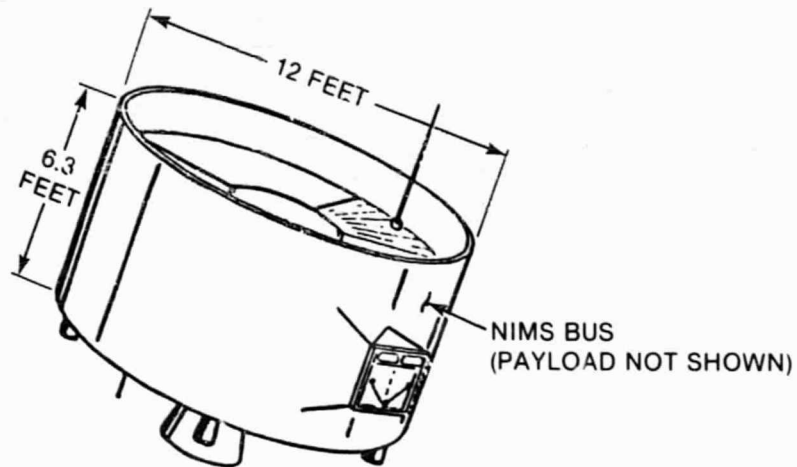FIGURE 2 Schematic diagram of the Dynamic Isotope Power System (DIPS).

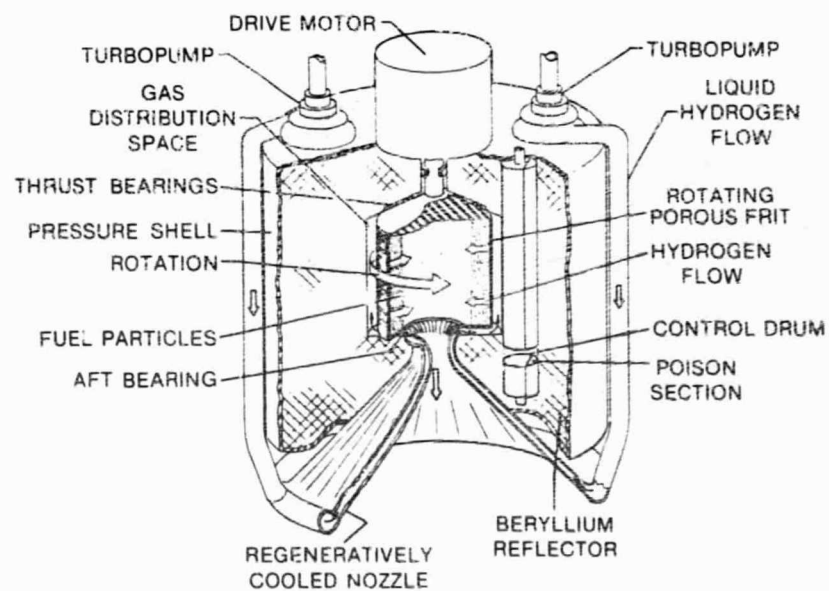FIGURE 3 Nuclear Integrated Multimission Spacecraft (NIMS) concept, incorporating DIPS.

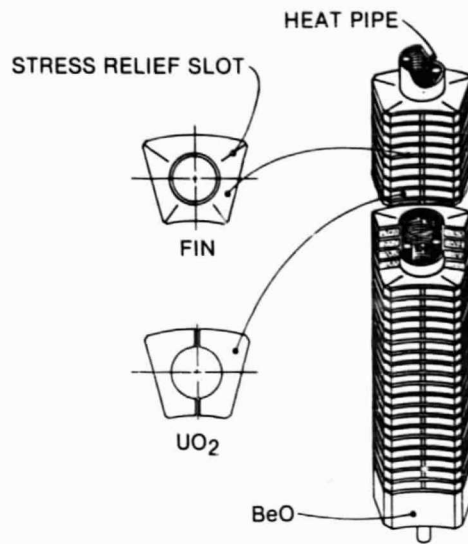FIGURE 4 Rotating Fludized Bed Reactor concept (early rocket form).

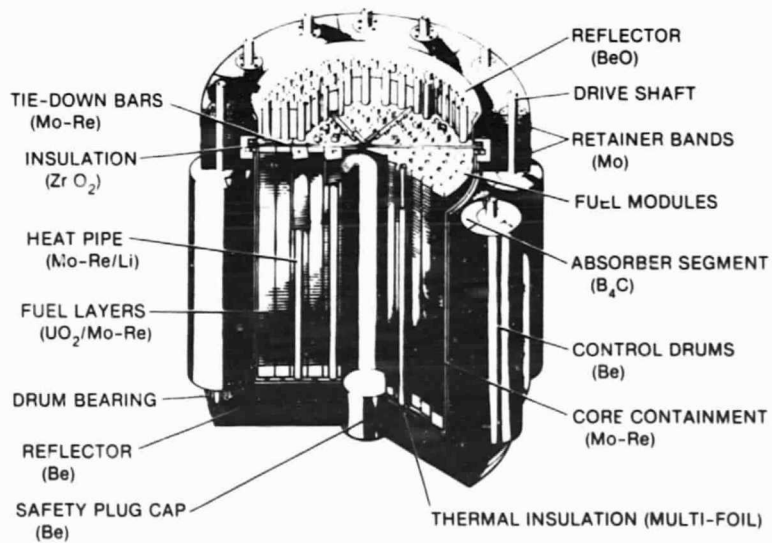FIGURE 5 Heat pipe fuel element module, SP-100 reactor.

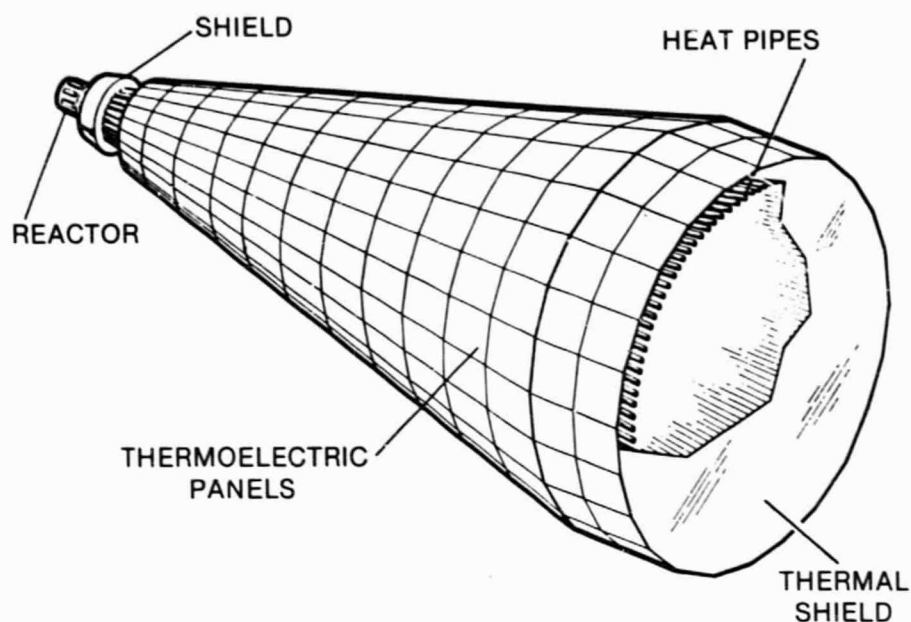FIGURE 6 SP-100 nuclear reactor power system.
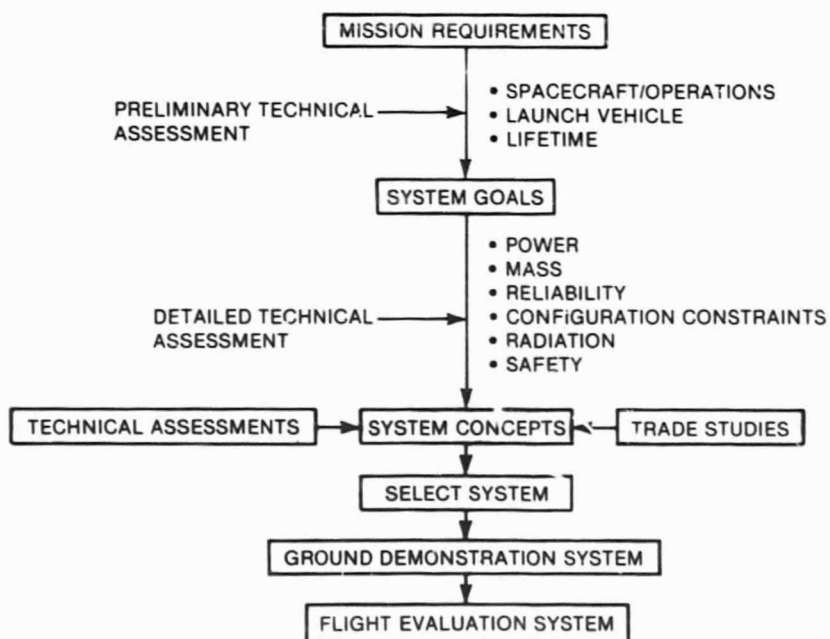
FIGURE 7 SP-100 nuclear reactor power system.

FIGURE 8 Basic steps in system planning and development.

MEETING FUTURE SPACE ENERGY NEEDS

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In the past, space energy needs have been relatively modest and have
been met by well-understood but expensive technology such as primary
fuel cells, solar arrays with batteries, and radioisotope
thermoelectric generators. As operations with the Space Shuttle
expand, we face an age when the cost and difficulty of supplying
energy could very well become a practical limit to mankind's endeavors
in space. This paper discusses a number of possible applications of
high energy in space and explores the potential role of nuclear
reactor power systems in satisfying such needs. Since none are
requirements at present, they should be viewed as justification for
technology development, not for system development.

Figure 1 illustrates the space energy demand over the short history
of the U.S. space program since 1966. Energy, that is, power times
design mission duration, is plotted for each of several National
Aeronautics and Space Administration (NASA) and civil missions against
launch year. It is noted that for these past missions, total energy
requirements have been modest, with peak demand in the range of a few
tens of thousands of kilowatt-hours (kWh), and in all cases less than
50,000 kWh. In the following discussion, a number of possible future
missions are reviewed, each of which demands total energies much
greater than this. Many of these missions also require energy system
characteristics that are different from those of past systems and that
at this time appear to be best satisfied by the nuclear fission option.

We have pushed back frontiers in space with the early Earth orbital
missions and with missions to the Moon, Mars, Venus, and Mercury.
Recently, we have had success in the Pioneer and Voyager missions to
the giant outer planets Jupiter and Saturn. If the Voyager 2
spacecraft continues to function properly, we may expect to see our
first detailed views of Uranus in 1986, and possibly even of Neptune
in 1989, some 12 years after the launch of the spacecraft. Such
missions are bold precursors, leaving many important scientific issues
unresolved even after years of exploration. For example, given the
planned Galileo mission to Jupiter, we know that a number of questions
will remain about the details of that body. As for distant Saturn,
while a spacecraft similar to Galileo could characterize certain ring

dynamics, it is likely that several questions regarding Saturn's rings will still be unresolved. These questions deal with the origin and processes of ring formation and composition, as well as the nature of the ring particles themselves, and will likely remain until there are detailed observations of individual ring particles and in situ composition measurements of the particles and their immediate environment. The exploration of Saturn's rings is very important to the understanding of the creation and evolution of Saturn and its satellite system. While the rings are only a small part of the mass of the system, their existence must be accounted for in any complete theory.

A possible mission to rendezvous with Saturn's rings has been studied and is illustrated in Figure 2. This mission employs a nuclear-powered electric propulsion system that takes it from Earth to Saturn in about 8 years. Upon reaching the vicinity of Saturn, it spirals in, in the plane of the rings. Once reaching the outermost ring, it aligns its thrust vector slightly toward the ring plane, which boosts it above the ring plane in a non-Keplerian, minor circle orbit. Still spiraling in toward Saturn, the spacecraft moves above the rings at a distance of only 1-20 km. In about 285 days, the spacecraft reaches the innermost ring after having made detailed observations spanning the entire ring system. A ballistic mission, even without the ability to continually stay above the ring plane, would require about a quarter of a million kilograms of propellant to achieve an orbit inside the innermost rings. Nuclear power and electric propulsion thus would make this mission possible. It is the only practical way it can be done. It must be noted that such a mission demands an energy expenditure of 5 million kWh.

Missions to the far outer planets--Uranus, Neptune, and Pluto--have long flight times. Voyager 2 will reach Neptune 12 years after launch after completing gravity assists by the major bodies Jupiter, Saturn, and Uranus. Ballistic orbital missions to Neptune will take much longer because of the requirement for low orbit insertion energy, which in turn requires a low approach velocity. Just such a mission has been studied utilizing a Galileo-derived spacecraft that would launch in 1991 and utilize a Jupiter swingby, arriving at Neptune some 20 1/2 years after launch. Clearly, such missions would require more continuity in programming than we can anticipate at this time. However, the graph in Figure 3 illustrates that nuclear power in conjunction with electric propulsion may enable such interplanetary missions to be flown. Payload versus flight time to Neptune is shown for three levels of nuclear system power per unit mass--specifically, 40, 30, and 20 W/kg. A reference mission of this type that has been studied in some detail can achieve higher net payload for a flight time of about 12 years than can the ballistic Galileo type mission. In general, it can be stated that nuclear reactor technology combined with electric propulsion permits practical flight times for far outer planet orbital missions. The benefits of such flight time reduction are not limited to increased confidence in the reliability of the

spacecraft. The investment required to build such a vehicle can be postponed with no loss in results, or the benefit can be captured much sooner. The total energy required for such a mission is also of the order 5 million kWh.

Direct Broadcast Satellite (DBS) is an extension of current communication satellite technology. In one version, it enables terrestrial broadcast stations to transmit to a geosynchronous satellite relay system and directly link with the user by means of a home antenna 1 m in diameter or less. As evidence of the advances of this technology, the Federal Communications Commission (FCC) has recently adopted interim DBS services rules for the authorization of a variety of TV broadcast services in the continental United States (CONUS) and beyond. Thus one corporation, for example, will have permission to operate four satellites spaced 20° apart along the geostationary arc to service areas in the CONUS that are approximately the size of the time zones.

Economies of scale and congestion in geosynchronous orbit may eventually dictate consolidation of the direct broadcast system into fewer satelites than present technology permits. The Advanced Direct Broadcast Satellite (ADBS) depicted in Figure 4 represents a possible second or third-generation direct broadcast concept. The graph in Figure 4 illustrates the trade-off between geosynchronous antenna size and available power for ADBS, assuming fixed receiver characteristics. Nuclear space power should enhance the economic viability of ADBS by providing inexpensive, high transmitter power along with energy to electric propulsion engines for LEO-to-GEO (low earth orbit to geosynchronous earth orbit) transfer and station keeping. A nuclear reactor power system would permit operation of ADBS in optimum longitude bands currently unattractive owing to earth eclipse periods. (Near-term DBS satellites will operate in far western longitudes to ensure that eclipse periods occur after local midnight.) The concept illustrated would require some 1.3 million kWh for both broadcast and station keeping over a 10-year period.

Figure 5 illustrates a large manned orbital facility that could operate in LEO for space industrialization and utilization of resources in the next century. Such a facility could require capabilities well beyond those of the space station currently contemplated for the 1990s. Three key benefits of nuclear power systems for manned orbital facilities are reduction of the drag makeup energy requirements, reduction of the interactions of the power system with the attitude control system, and elimination of orientation requirements. For example, drag makeup propellant resupply requirements are proportional to the area of the manned orbital facility. The graph in Figure 5 illustrates the difference in area for a nuclear versus a solar array/battery system as a function of power level. A significant reduction in projected area accrues from the utilization of nuclear power systems, which in turn reduces drag and the amount of propellant needed for orbital altitude maintenance.

It is difficult to estimate total energy requirements for such a future facility, but some studies have suggested demands as high as 20 million kWh for an early 21st century undertaking.

On the earth it has been found to be economical to generate power at a central station and distribute it over transmission lines to the user. Should a space power system be available that offers such economies of scale in contrast to solar arrays, a central power station in space may be practical. Studies of this concept that have been done to date have suggested that very high demand scenarios must exist before economic benefit will result. In technology programs, we have begun to address power transmission in space. Such a concept would eliminate the requirement for each spacecraft to maintain a complete power generation system, carrying in its stead an energy receiver. Figure 6 illustrates this concept, and the inset shows the dependence of required antenna size versus transmission distance on the wavelength of the radiated energy. Both microwave and laser systems appear to be candidates, but the required high efficiencies have to date been demonstrated only for microwave systems. The central power station depicted requires an energy of 10 billion kWh.

Recent studies projecting to the year 2000 show a tenfold to hundredfold increase in annual spacecraft mass delivered to geosynchronous orbit. Propellant requirements for chemical tugs may become sufficient to justify a nuclear-powered orbit transfer vehicle (OTV) even though such an approach carries with it substantial trip time penalty.

Figure 7 shows payload in geosynchronous orbit versus one-way trip time of a round-trip nuclear tug mission. These data are presented for varying levels of technology. The conservative curve shows flight time versus payload capability for nuclear power technology based on thermoelectrics operating at a hot junction temperature of 1275 K with a figure of merit of 0.7×10^{-3} K^{-1}; the improved curve assumes 1 $\times 10^{-3}$ K^{-1} and 1350 K, and the advance curve assumes 1.4×10^{-3} K^{-1} and 1525 K. The points along the curves show optimum power required for the best electric thrust subsystem needed for each combination of payload and flight time to GEO.

Nuclear performance can be compared to chemical OTVs envisioned in the future. For example, a 100-kW nuclear electric propulsion (NEP) tug based on near-term technology can place a 5,600-kg payload in geosynchronous orbit in about 260 days. A chemical tug could place the same payload in GEO in a few hours, but it requires the shuttle to deliver 16,000 kg of propellant. For the same shuttle cargo mass of 21,600 kg, a nuclear tug could deliver two payloads instead of one. Were this capacity sold to another user, it would increase the economic efficiency of operating the shuttle by a factor of 2. Provided flight time penalty is acceptable, a nuclear tug could therefore serve to either increase payload capability or improve the economic efficiency of the shuttle. For a lifetime of eight round trips such a tug would require of the order of 6 million kWh.

As indicated earlier, by the year 2000, the mass of equipment being placed in geostationary orbits annually may be 10-100 times the current level, making space transportation a multibillion-dollar industry. Assuming a continued reliance upon hydrogen/oxygen upper stages, nearly 75 percent of the mass launched from the earth is in the form of propellant, with perhaps 300 t/yr of oxygen. In terms of energy required for transport, the moon is 10 times closer to LEO than the earth's surface. As indicated in Figure 8, studies are under way of the feasibility of processing lunar material into oxygen for use for eventual transportation and use at LEO. These studies indicate the need for high electrical power, from 100 kW to several megawatts, and assume the use of photovoltaic technology. However, because of the 2-week lunar dark period the duty cycle is only 50 percent for this system. A nuclear reactor system could increase this to 100 percent. This is only the most recent scenario suggesting the desirability of a lunar base; there are, of course, many others that invoke manufacturing or exploration as motivations. In any event, there is no doubt that such a venture would demand very high energy requirements: in one recent estimate, 1 billion kWh.

In Figure 9, the energy requirements for these few missions are overlaid on the historical basis previously displayed in Figure 1. It is evident from the orders of magnitude involved that supplying such demand will be an enormous challenge. It is equally evident that it is extremely important to examine alternative system approaches in preparing for the future. If terrestrial commercial energy costs of roughly 10 cents/kWh were attainable on the moon, a lunar base would cost around $10 million/yr to operate. At the current space cost of about $1,000/kWh shown in Figure 10, the bill would be over $100 billion/yr for operations. For such missions as we have been discussing to be at all practical, much lower unit energy costs must be achieved. Technological change as well as economies in scale must be invoked. When it is realized that the total power installed in space to date for civil missions is of the order of 100 kW, the need for practical alternatives to meet the demands of this new age are sharply reinforced. The nuclear option is clearly one such alternatives.

Besides offering cost-effectiveness and permitting operations to be carried out in the farthermost regions of the Solar System, nuclear space power makes possible operations in some hazardous environments such as the earth's Van Allen belts. Figure 11 shows power per unit mass for nuclear and solar power systems operating for 10 years as a function of earth orbit altitude. It illustrates the effect of the earth's radiation belts on the specific power of solar energy storage systems for levels of technology ranging from solid substrate silicon arrays with nickel cadmium batteries to advanced flexible arrays with regenerable fuel cell storage. The nuclear system has constant specific power throughout this altitude range. The nuclear specific power band covers the ranges of technology discussed previously. The benefits of nuclear power are clear for this case. It will permit

more effective operation in a region of near-earth space. The mass advantage suggested for nuclear systems on this chart is expected to become even more pronounced at higher power levels.

Figure 12 illustrates some of the technology issues that must be resolved if nuclear reactor power systems are to be considered for practical use in space. Beyond the reasonably well understood nuclear reactor, heat transport, safety, and associated radiation shielding technological issues are a broad range of critical thermal-to-electric conversion and system questions that require research and technology efforts. For example, regarding energy conversion, research is needed on advanced thermoelectric materials as well as on alternative approaches that would permit higher efficiencies at lower temperature. The achievement of lightweight thermal control may require the use of materials such as composites whose properties in such applications are not adequately understood. New heat rejection concepts have been suggested that could dramatically influence overall system approaches, and they should be investigated. Dynamics of the large bodies to be flown are likely to demand advances in distributed active control. There are serious concerns about ineractions with natural or induced environments--ranging from vehicle charging to radiation to space plasma effects--that demand code development to permit adequate prediction and design criteria to be established. Power management techniques and high-voltage components, such as transistors, capacitors, cabling, and connectors, need to be made available at power levels and radiation hardness levels consistent with expected environments. Compatibility with shuttle and manned operations needs to be established. A comprehensive technology program must be structured so as to resolve this broad range of issues in the context of possible applications.

To summarize, nuclear reactor space power systems have the potential of being less massive for high-power applications than solar array systems, of being more capable of surviving penetrating high radiation environments, and of enabling missions to be launched to the outer reaches of the Solar System. Nuclear space power systems also offer the potential of being less costly at high power levels than some alternative forms of space energy.

It is clear that realization of the promise of space depends largely on the availability of abundant, relatively low cost energy. Nuclear space power appears to be one option that could meet future demands at lower cost for high power levels and as such is deserving of further study. Now is the appropriate time to advance the technology base to permit a more complete understanding of the nuclear space power option.
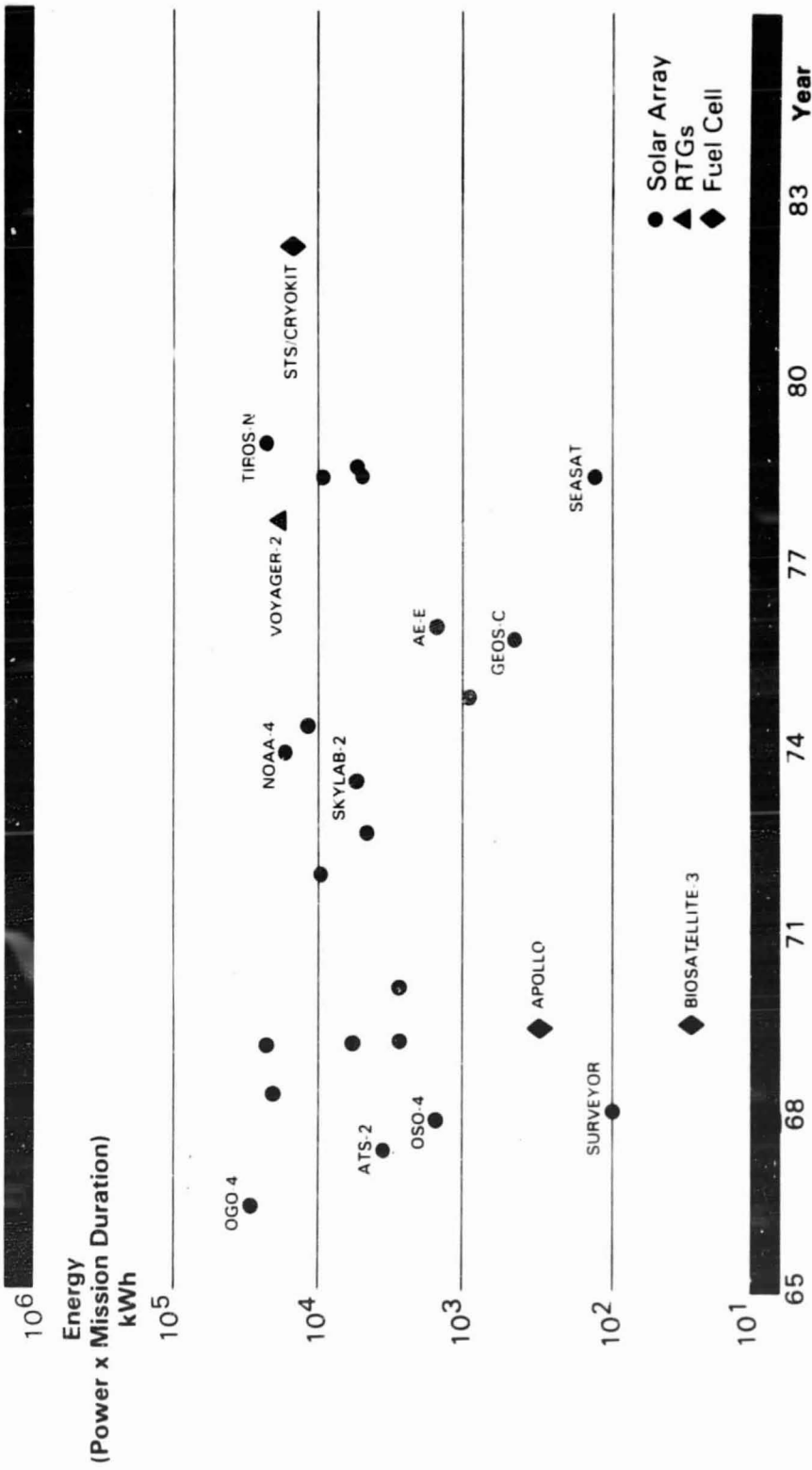
FIGURE 1 Space energy demand history.

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5 x 10^6 kWh

■ *Resolve Nature of Rings*

■ *Spiral 1-20 Km Above Ring Plane*

■ *10 Year Mission to Inner Rings*

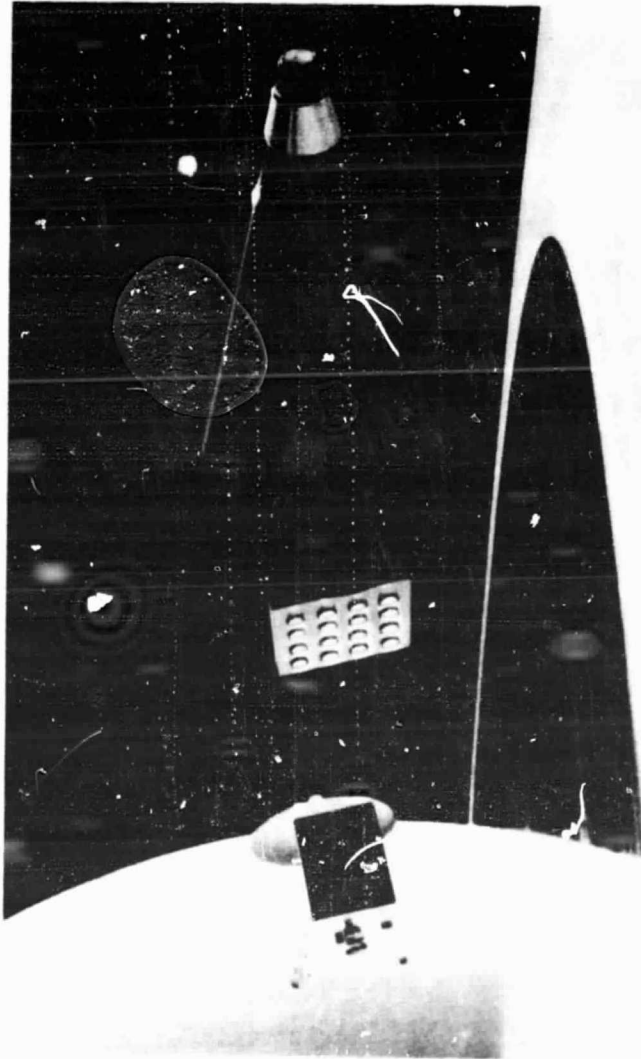■ *Nuclear Enables This Mission*

FIGURE 2   Rendezvous with Saturn's rings.
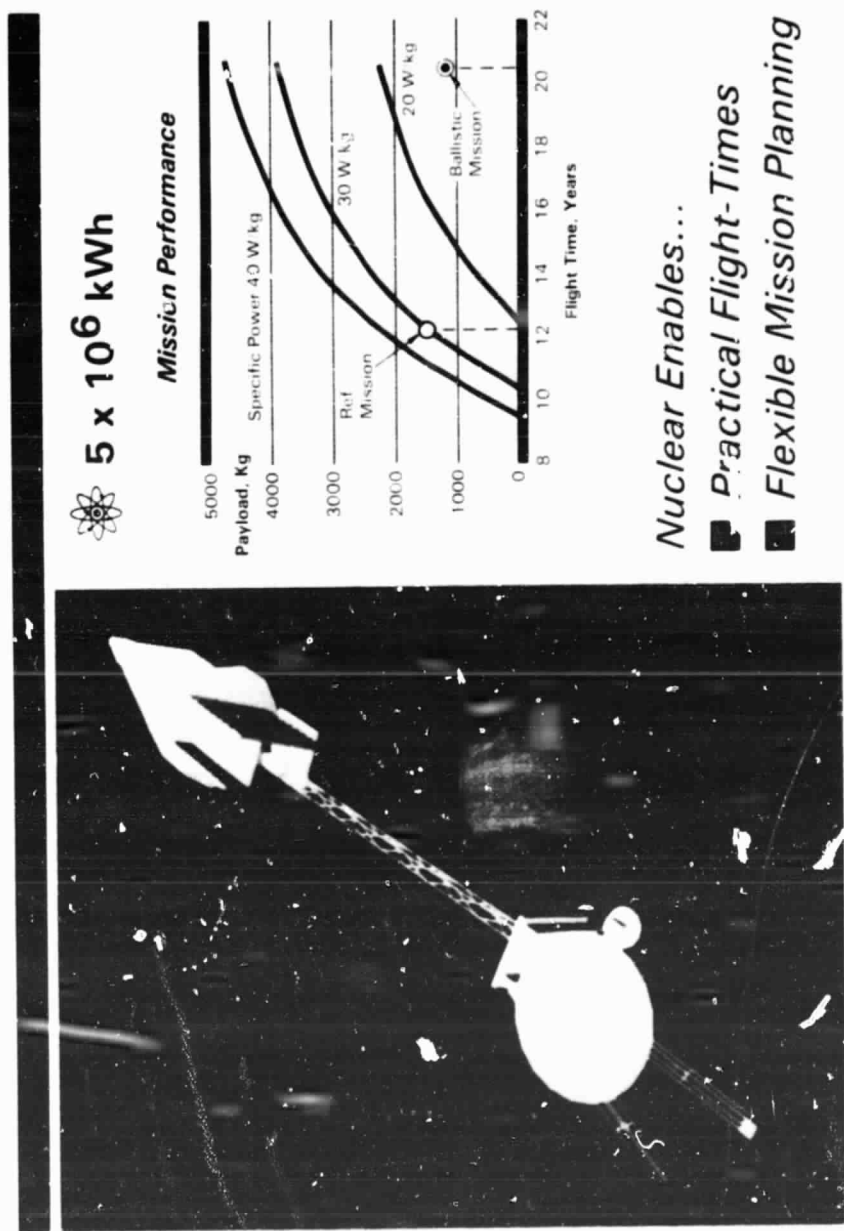
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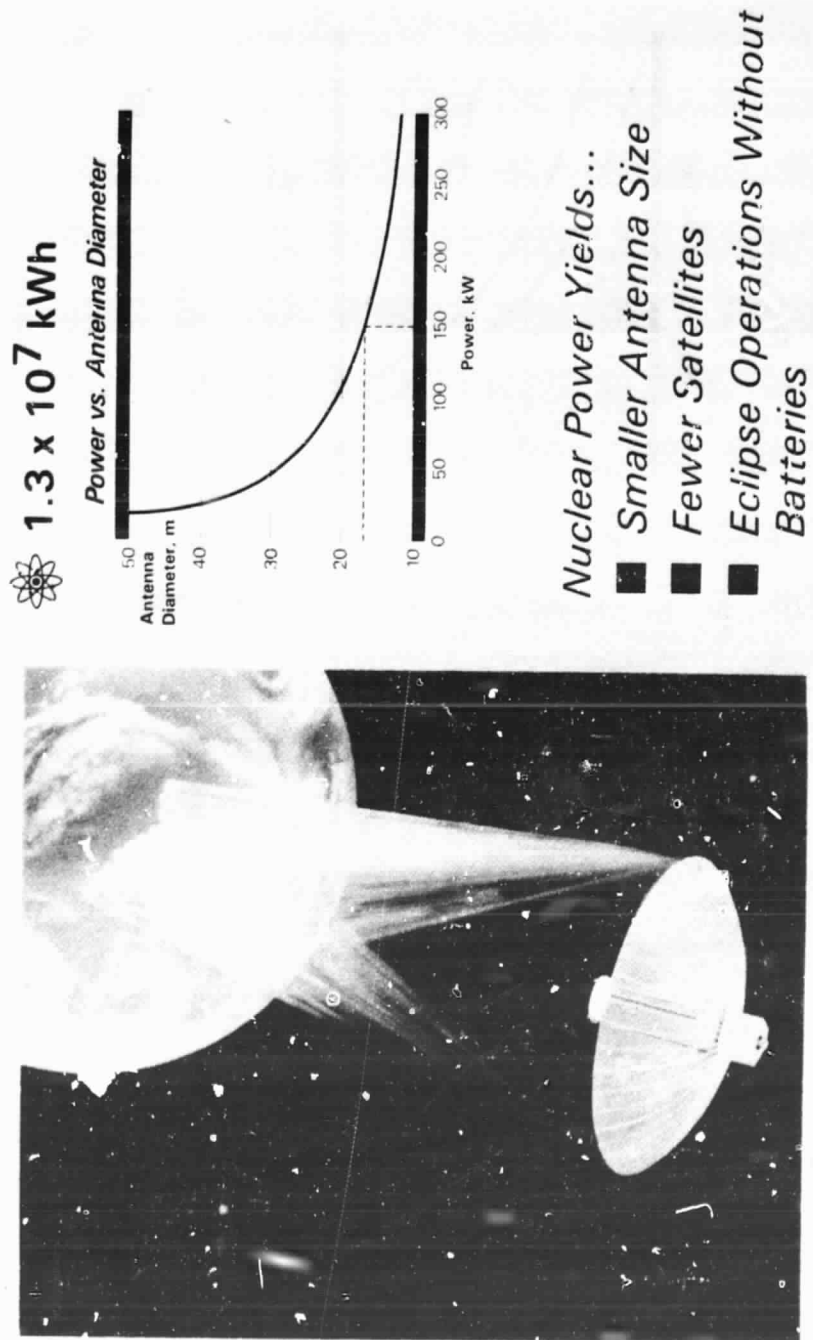FIGURE 3   Neptune orbiter mission.
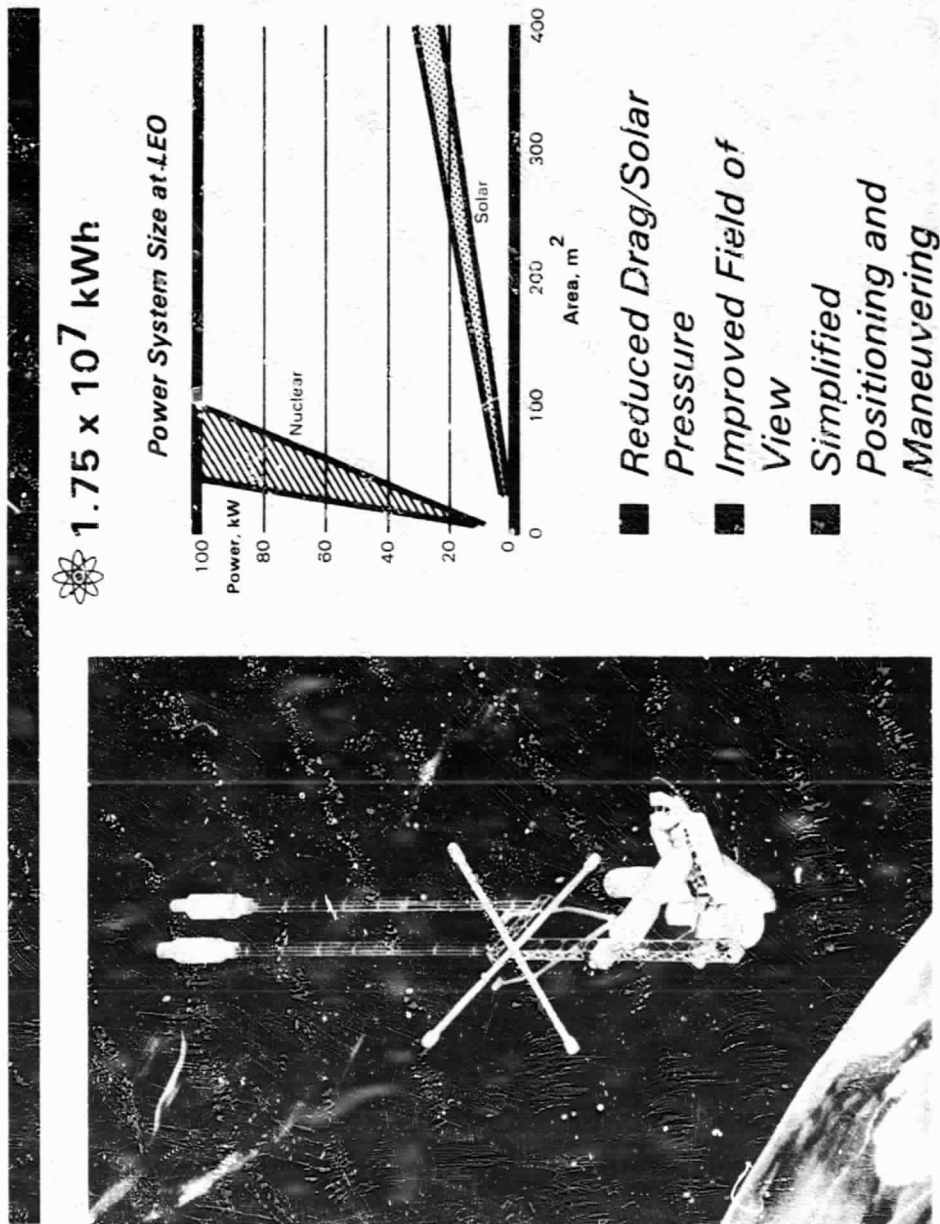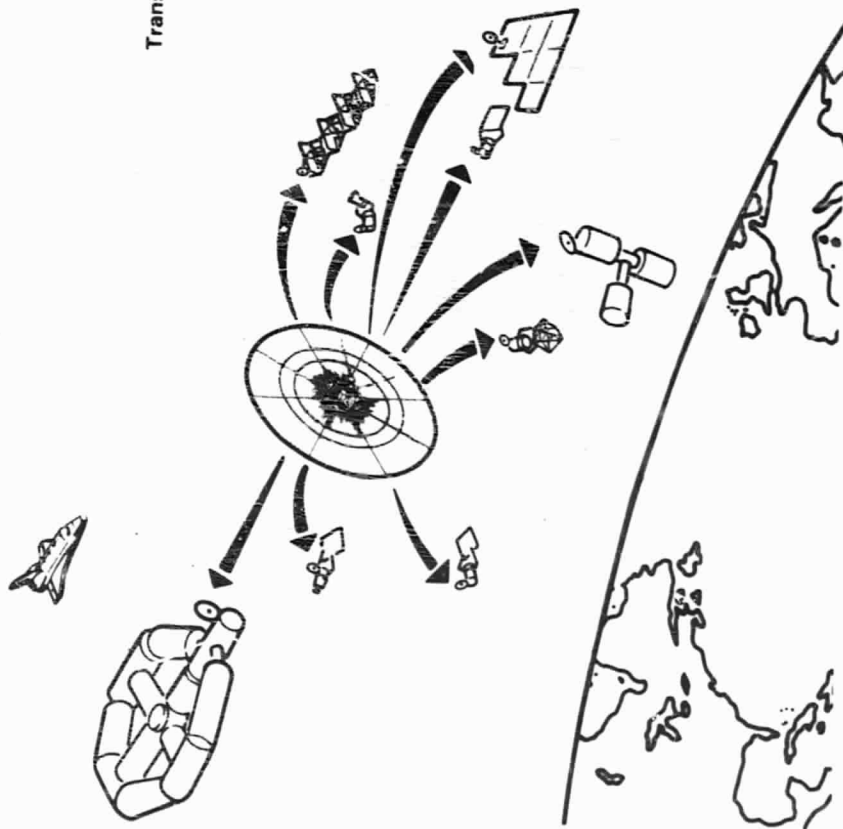
FIGURE 4   Advanced Direct Broadcast Satellite.
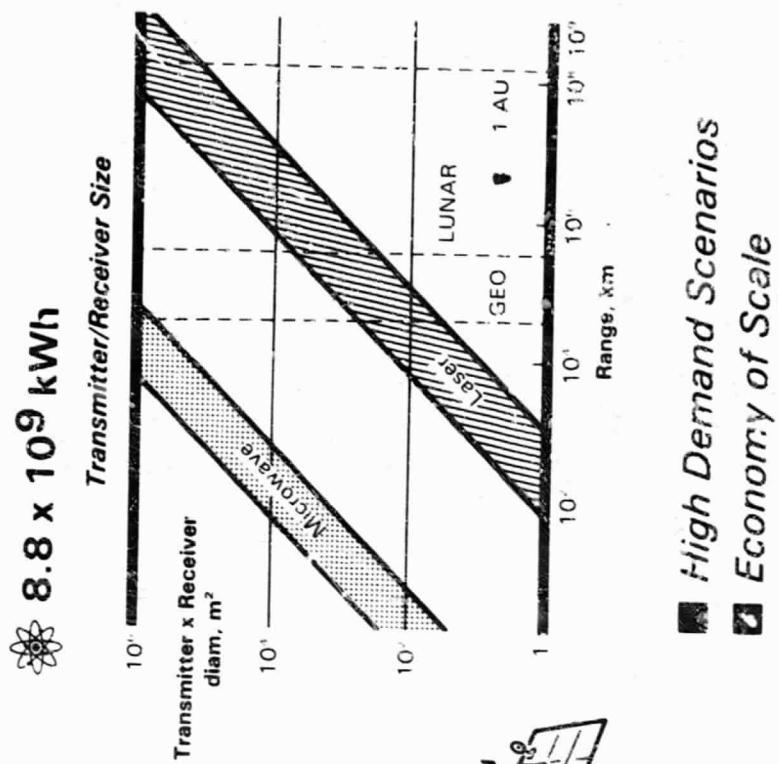
FIGURE 5   Manned orbital facility.
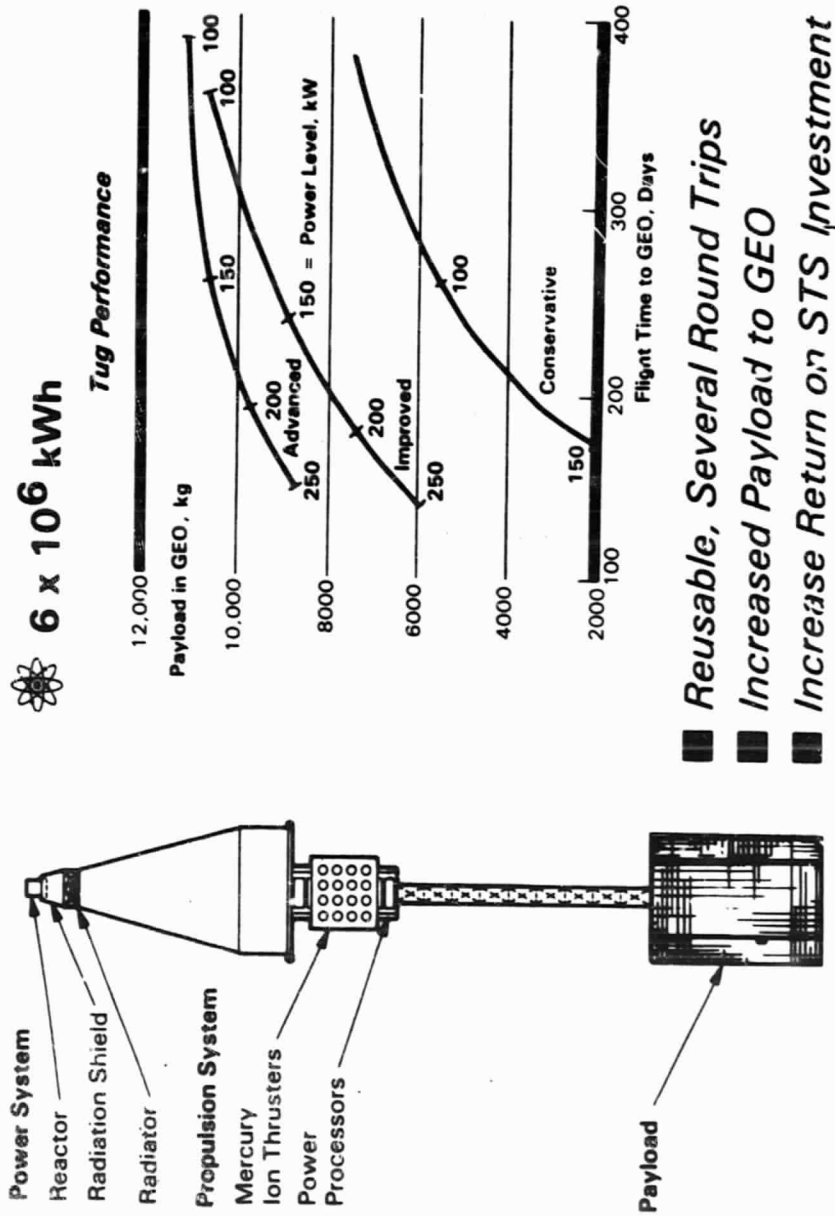
FIGURE 6 Central power distribution.

FIGURE 7 Nuclear tug.

FIGURE 8 Future lunar base.

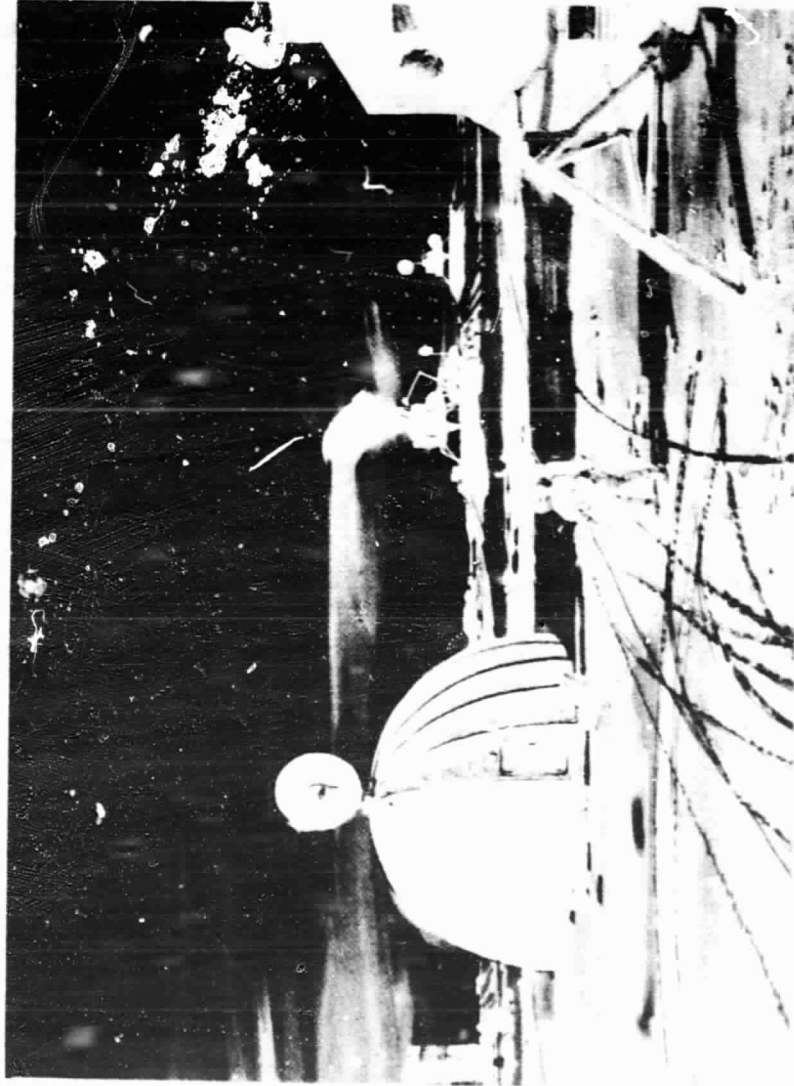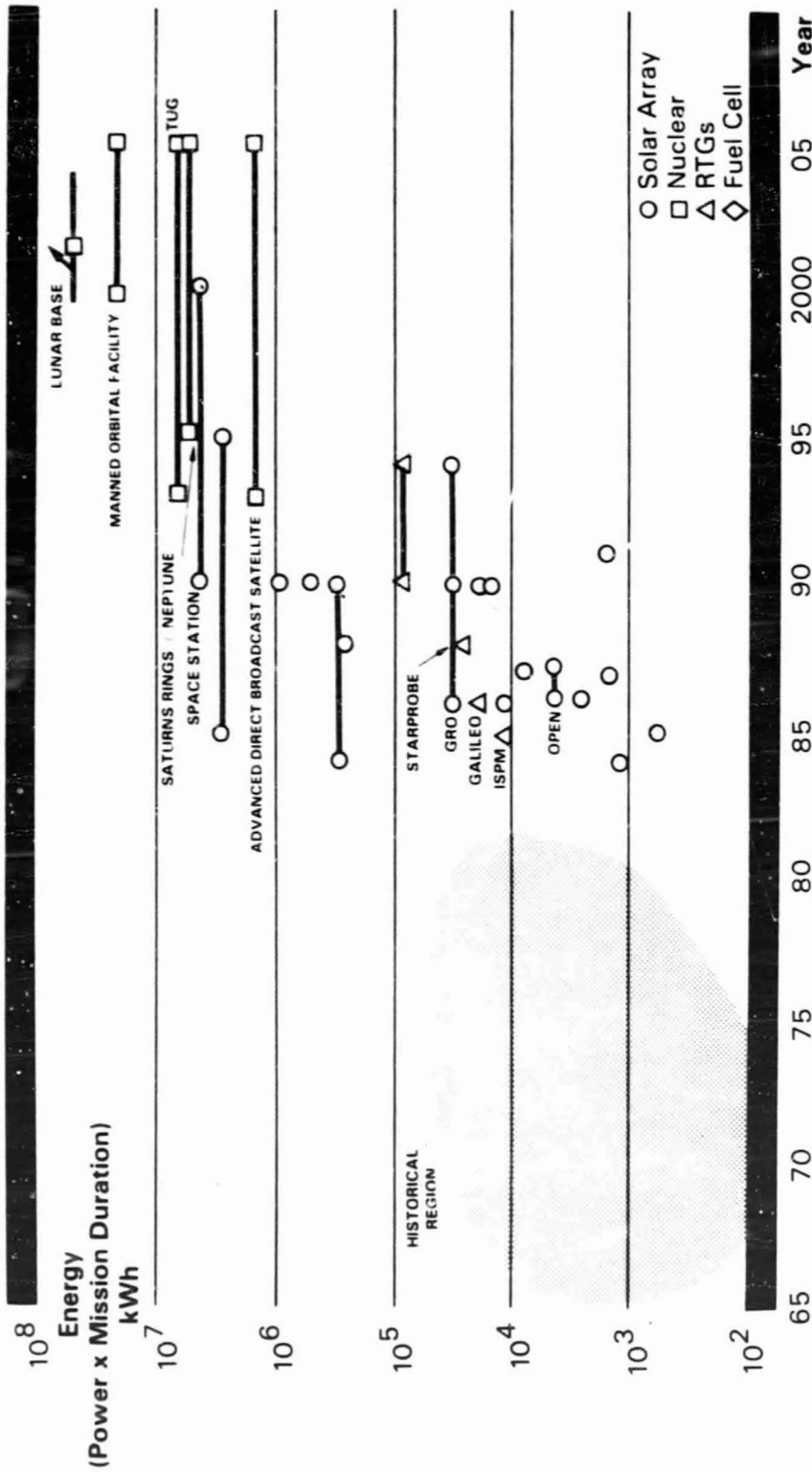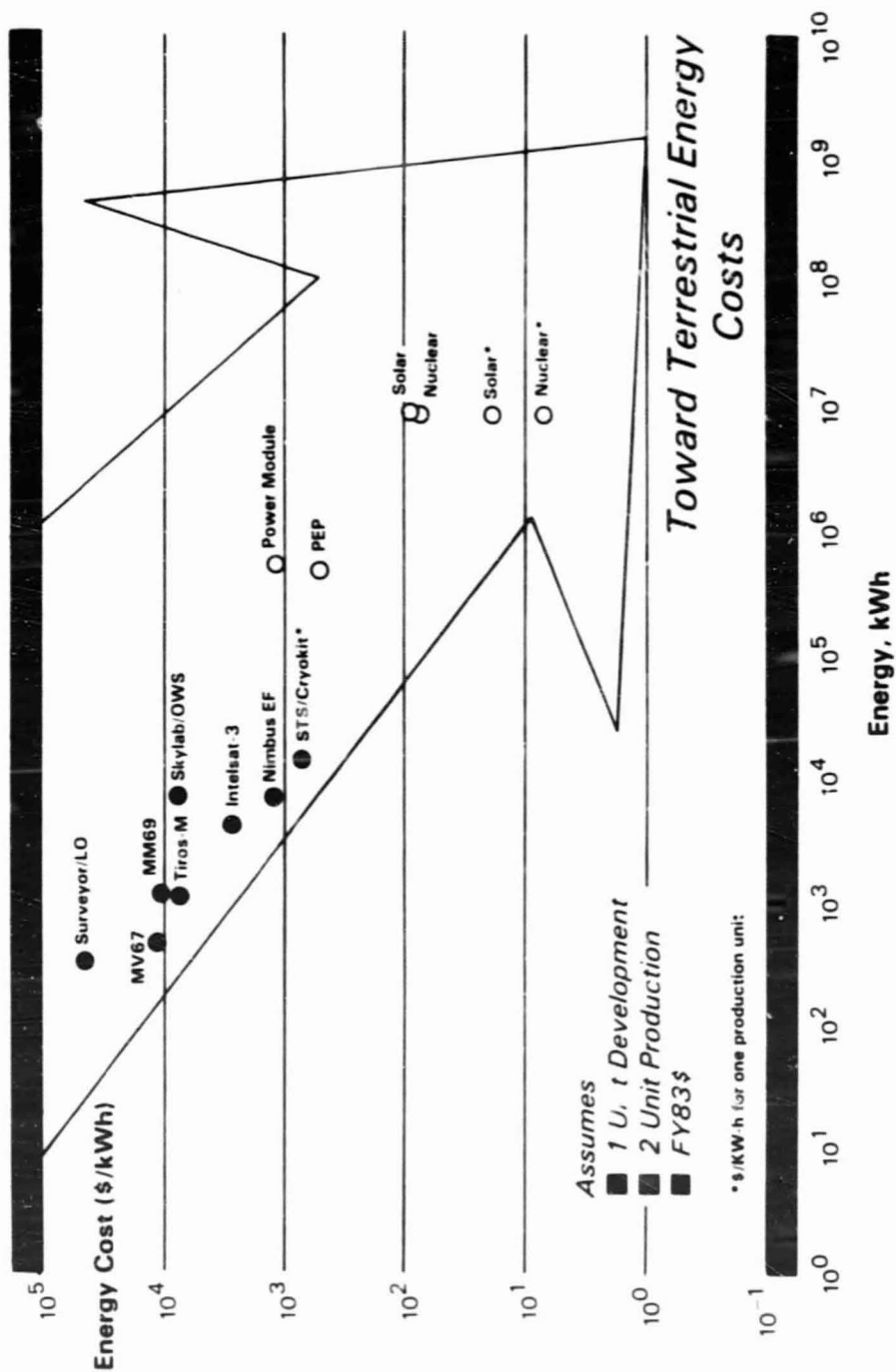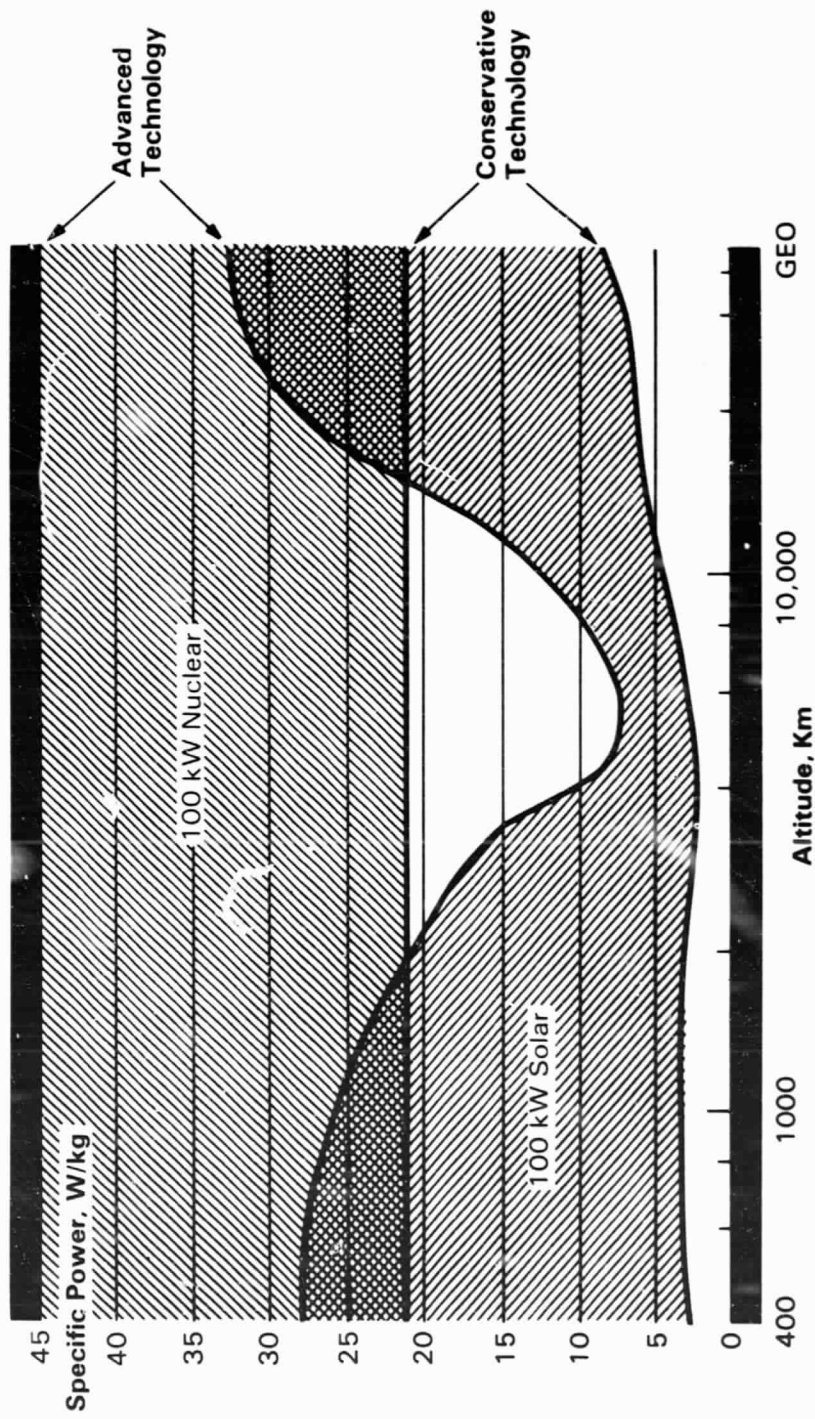FIGURE 9   Future space energy demands.

FIGURE 10   Space energy costs.

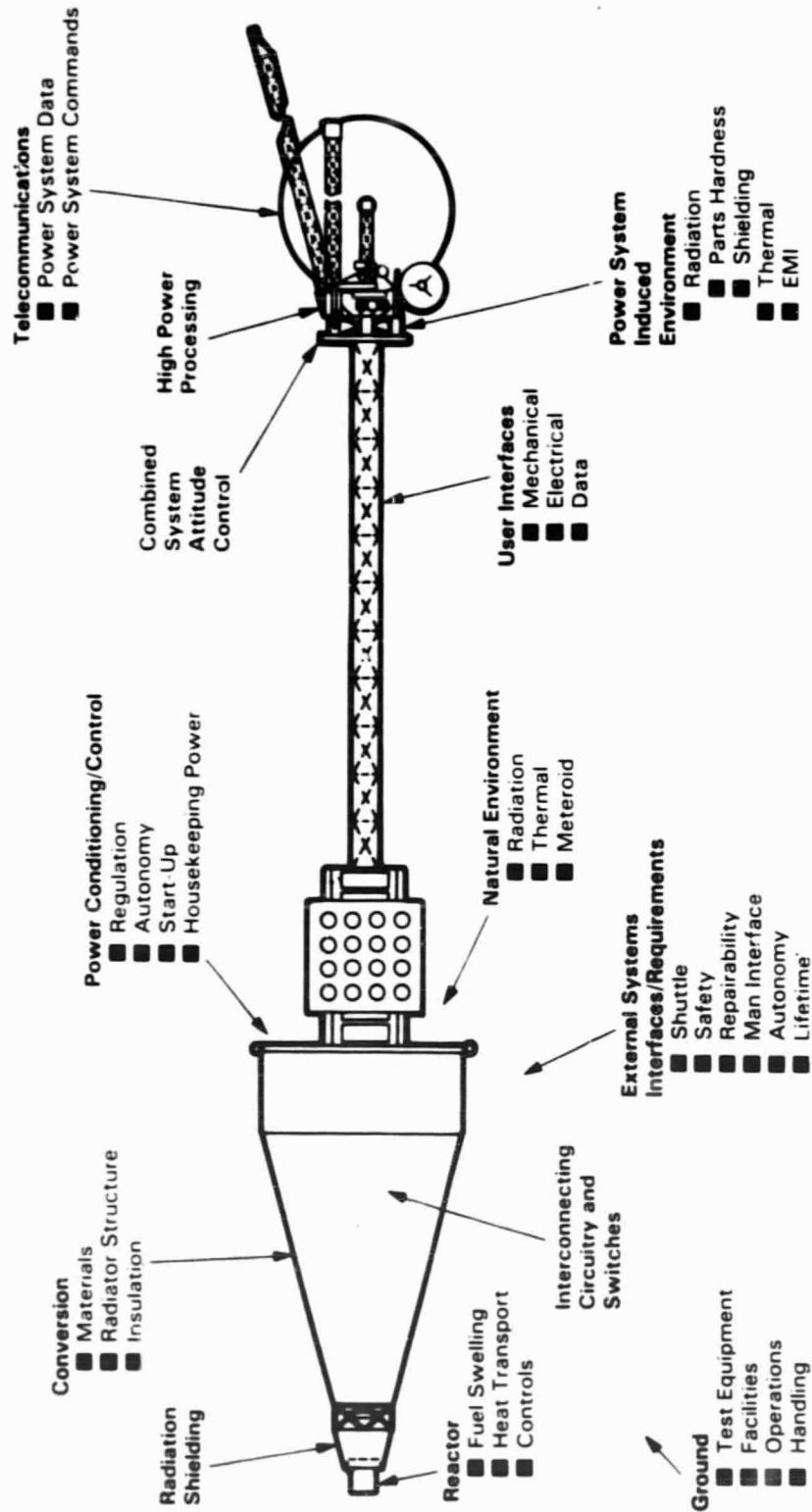FIGURE 11 Long-term operations in Van Allen belts: Possible with nuclear space power.

FIGURE 12 Nuclear space power technical issues.

SPACE NUCLEAR REACTOR POWER APPLICATIONS

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INTRODUCTION

A review of projected space power system design requirements for the
time period to 1995 has been performed with the view of evaluating the
impact of these requirements on the potential use of nuclear reactor
power systems and alternative solar photovoltaic power sources.
Emphasis was placed on attempting to define the space missions that
would dictate the utilization of nuclear reactor power systems. In
addition, the constraints and concerns the reactor system user would
then have to deal with were addressed. The review includes the space-
craft military threat environments as well as the projected power
system performance requirements in order to consider as many appro-
priate power system selection factors as possible.

SPACE MISSION POWER REQUIREMENT

Projections of near- and long-term requirements for space power are
found in several sources. Table 1 contains a representative list of
these sources. Representative papers discussing space power projected
technology advances against projections of space power requirements
are those by Barthelemy and Honneywell (1982) and Mullin et al.
(1982). It is noted that the requirements are defined by the purpose
of the proposed spacecraft mission and that the candidate design
solutions are proposed by various government and industry elements
specializing in the applicable technology. Table 2 is a list of
representative sources of advanced space power system capabilities,
either demonstrated or forecast. The military space system missions
can be classified as shown in Table 3.

In addition to power system characteristics, the projected military
space missions will be exposed to a variety of physical threats. The
principal threats are identified in Table 4. These threats and their
impact on candidate power systems are discussed briefly below. An
unambiguous definition of threats and levels of protection is not

TABLE 1 Sources of Space Power Requirements Definition

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1. Air Force technical objectives documents

2. Requests for proposals for space systems (NASA and DOD)

3. NASA- and DOD-prepared papers in:

   American Institute of Aeronautics and Astronautics (AIAA)
      proceedings

   IECEC proceedings

   IEEE Photovoltaics Specialist Conferences

4. NASA and DOD participation in AIAA technical committees

5. NASA and DOD space system technology workshops

   Military space system technology workshops

   NASA-OAST presentation at workshops

6. Contractor proposals for advanced space systems in the open
   literature.

7. NASA and DOD participation in technical conferences, such as the
   EIA Space Electronics Conference.

NOTE: NASA, National Aeronautics and Space Administration; DOD,
Department of Defense; AIAA, American Institute of Aeronautics and
Astronautics; IECEC, Intersociety Energy Conversion Engineering
Conference; IEEE, Institute of Electrical and Electronics Engineers;
NASA-OAST, NASA Office of Aeronautics and Space Technology; EIA,
Electronics Industries Association.

practical at this time, because some threats are only now evolving.
The definition effort, however, is ongoing.

Nuclear Threat

The Joint Chiefs of Staff (JCS) Guidelines have provided criteria for
hardening levels in satellite designs. Reference levels for the X-ray
environments have been established. The other components of the
nuclear threat are the gamma-ray dose, the neutron fluence, the

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TABLE 2 Sources of Advanced Space Power System Capabilities

_____

1. Reports on technology development studies and testing

    Contractors (government funded)

    NASA center

    DOD laboratories

    National laboratories

    Universities

    DOE

2. Contractor and government briefings on funded and in-house
    technology development

3. NASA and DOD space system technology workshops

4. Papers in:

    IECEC proceedings

    IEEE Photovoltaics Specialist Conference proceedings

    EIA Space Electronics Conferences

    DARPA Strategic Space Symposia

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NOTE: NASA, National Aeronautics and Space Administration; DOD,
Department of Defense; DOE, Department of Energy; IECEC,
Intersociety Energy Conversion Engineering Conference; IEEE,
Institute of Electrical and Electronics Engineers; EIA,
Electronics Industries Association; DARPA, Defense Advanced
Research Projects Agency.

system-generated electromagnetic pulse (SGEMP) associated with the
above radiation, the dispersed electromagnetic pulse (DEMP), and
debris-induced gamma rays. Trapped fission electron hardening is
provided for in these guidelines. Since 12 cal/g will melt solder,
the use of solar cell welding for array assembly has been adopt   for
hardened designs. The neutron damage is principally in the form of
atomic displacement, resulting in some degradation in solar cell
performance. SGEMP is the collection of X-ray-induced interactions

TABLE 3 Military Space System Missions

Communications (10-100 kW)

     Laser communication with submarines

     Space-based electronic jammers

     Real-time battle communications

     Control of remotely piloted vehicles

Navigation (1-10 kW)

Weather (1-10 kW)

     Real-time weather display for aircraft

     Tactical support

Surveillance (10-100 kW)

     Space-based radar to track ships, missiles, aircraft, and satellites

     Infrared trackers for missiles, aircraft, and satellites

Directed-energy weapons (10-100 MW)

     High-energy lasers

     Particle beams

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and effects that create a localized electromagnetic environment. This environment can induce unwanted signals in electronics. The solution of problems caused by SGEMP is complex, but better analysis and modeling techniques are being devised.

DEMP results from the interaction of an electromagnetic pulse from a high-altitude exoatmospheric nuclear detonation that is propagated through the ionosphere to a satellite. The geometry of the threat limits its significant effects to satellites with their lines of sight through the ionosphere to the burst point. Solar arrays require circuit protection techniques to withstand EMP, usually at the cost of a weight penalty.

TABLE 4 Satellite Threats

1. Space-based, airborne, and ground-based lasers

2. Space-based particle beams

3. Nuclear weapons

    Gamma rays

    Neutrons

    Debris

    X-rays

    Electron effects, trapped electrons, SGEMP, DEMP

4. Nonnuclear antisatellite warfare

Trapped electrons are those contained in the earth's magnetic fields. These fluences can be enhanced by the addition of high-energy fission electrons (200 keV and above). The enhanced electron environment is long lasting and can result in doubling the dose to satel ites at sensitive altitudes. In addition, the electrons can create charge buildups in internal satellite dielectrics. If the charge exceeds the dielectric breakdown strength of the material, a discharge may occur, possibly burning out electronics components. Spacecraft charging can be caused by low-energy ambient electrons interacting with dielectric surfaces such as paints, thermal control blankets, and solar cell cover slides. The general threat, electron-caused EMP (ECEMP), is not yet well understood or defined.

Debris is the ionized material created from the vaporized materials and casing of the weapon. Above a few hundred kilometers in altitude, the debris is not contained by the residual atmosphere and magnetic fields but is ionized and may collect on a negatively charged satellite. Dose rate effects, rather than total dose, lasting a few hours, are possible when radioactive debris plates out on the surface of a satellite.

The hardening of solar array power systems for the nuclear threat is somewhat expensive and limited with present materials, but significant threat levels can be survived. The reactor power system is designed with materials that can withstand nuclear radiations, and the power-generating element is not as susceptible to nuclear radiation as the solar cell. The reactor system radiator thermal-optical performance, after the deposition of the X-ray energy,

must be kept at acceptable levels. It is assumed that hardening of
electrical and electronic systems in the two power sources would
present the same problems.

Laser Threat

A large number of survivability techniques have been defined for the
laser threat. It is apparent that the designer has a significant
number of options in solving the heat rejection problem associated
with certain levels of laser threat; however, with a solar array, the
available options are quite limited owing to the function of the array
and the materials available to achieve this function. Laser threat
hardening penalties for solar power systems may exceed 50 percent
under some engagement scenarios.

A balanced application of survivability levels to all satellite
subsystems is required, as a highly hardened element of the satellite
compared to a lower requirement for another component would not be
cost-effective. The trend is toward higher survivability levels.
Furthermore, some potential satellite maneuvering requirements can be
in the 0.001-0.1 "g" range, those in the upper end of this range
causing significant weight penaltie for solar arrays.

POWER SOURCES FOR MISSION CONCEPTS

The technology status and projected advances for the solar
array/battery space power systems are described, for example, in
Koenig and Ranken (1982), Lockheed Missiles and Space Co., Inc.,
(1982), Mahefkey (1982), Malbandian and French (1982), Mullin et al.
(1982), and Scott-Monck (1982). The projected technology for solar
power systems is shown in Figure 1; that for reactor power systems is
shown in Figure 2, and papers by Buden and Angelo (1982) and Powell
and Botts (1982) support this projection. The two curves in Figure 2
represent projections based on low and high funding levels affecting
the time at which a particular performance level is predicted to be
achieved. The predicted reactor power system performance in both
specific power and power level versus time is decidedly better than
that expected for the solar array/battery system.

Nuclear fission reactors for space power saw considerable
development in the 1960s, particularly the SNAP-10A (500 W(e)), which
was flight tested, and the SNAP-8, whose development was terminated
when firm missions could not be identified. The zirconium-hydride-
enriched ^{235}U fuel used in these systems can be considered state of
the art. Development of reactor systems for space power since 1973
has been limited largely to component fabrication and to design
studies being performed at Los Alamos National Laboratory and
Brookhaven National Laboratory. The current advanced reactor designs
utilize Mo-UO$_2$ fuel and would allow higher reactor operating

temperatures. National Aeronautics and Space Administration and
Department of Energy (NASA/DOE) design studies utilize thermoelectric
power conversion (9 percent efficiency) and can attain 25 W/lb at the
100-kW(e) design power level for electronics able to withstand 10^6
rads of radiation. A ground demonstration system could be completed
by 1990 by spending approximately $66 million. The Air Force/DOE
(AF/DOE) studies have been completed for a reactor that utilizes
thermionic power conversion (12 percent efficiency) and would attain
50 W/lb at the 1,000-kW(e)) design power level. Because of the long
development times and high development costs, a national space reactor
development program appears to be required to produce a flight
system. This would also solve the following dilemma:

1. A reactor power system does not exist, so use is not planned.
2. Use is not planned, so a system will not be developed.

The applications of reactor power systems to missions requiring 10
kW(e) to 1 MW(e) are feasible. Although of the missions studied in
this evaluation, several would have benefited from the use of a
reactor power system, its application was not mandatory and is not
expected to be mandatory until the 1990s because an alternative solar
array/battery power system was adequate in all cases. It is clear,
however, that a certain set of military mission requirements, if
established in the near future, could drive a development program for
technology readiness prior to 1990. The cost of such a development is
seen to be relatively high and appears to cause mission planners to
select nonreactor power systems in the interim. An early program to
accelerate the pace of reactor power system technology development is
encouraged so that the possibility of encountering a large technology
gap is minimized when firm requirements are identified. From the
evaluation of the missions, the 100-kW(e) to megawatt missions require
full consideration of nuclear power.

REACTOR USER CONSTRAINTS AND CONCERNS

The space program using a reactor power system has to accommodate
certain constraints and concerns associated with the nature of the
power system. Assuming an unmanned mission, the areas of concern are
structural, thermal, nuclear radiation, and safety. The structural
integration impact depends largely on the configuration of the reactor
power system. The power system with its thermal radiator may be a
unit that can be interfaced with the spacecraft at a given interface
plane either prior to launch or after the spacecraft and power system
are separately launched. If the mission does not allow this, then the
integration of the heat rejection radiator with the spacecraft may be
more complicated. The total integration may still be accomplished
prior to launch, or the heat rejection system may be treated as a
component separate from the reactor. In this case, the space radiator

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may be launched with the spacecraft, and the separately launched
reactor system hooked up in orbit. Figure 3 illustrates these
concepts. One of the concepts shows the reactor placed inside an
existing chamber as opposed to being attached outside. The
significance of different integration concepts is the desirability of
having a reactor power system technology development program that is
not bound to a particular integrated configuration for as long as this
is practical in the program.

The radiation doses to spacecraft components from the reactor must
be considered along with the natural environment and nuclear weapons
threats. This involves parts selection and placement of components so
that distance and shadow shielding by tanks, boxes, and such items as
storage batteries can be used to protect more sensitive components.
Computer programs are set up to handle the geometry/shielding/dose
computations. Reactor radiation test facilities are expected to be
used for the verification of certain components where existing test
data are not sufficient.

The logic and rationale for control of the reactor configuration
and the instrumentation are required for decision making. The reactor
start-up, temporary shutdown, or final shutdown rationale is
determined prior to launch. The spacecraft will be required to
control the electrical power from the system and will interface with
some of the reactor controls, instrumentation, and status data for
power system configuration control.

Range and nuclear safety efforts are significantly increased with
reviews, documentation, and liaison with DOE, AF, and NASA safety
organizations. The uncontrolled insertion of reactivity is of major
concern under all possible nonnominal conditions on the launch pad,
during launch, and in orbit. The user will provide the launch system
and spacecraft system data required to assess the potential hazards
and their probability of occurrence. Systems and procedures will be
designed to control potential hazards. The Space Shuttle launch of
reactor systems is expected to require significant safety review
activities. For manned missions, the personnel radiation shielding
and the operations controlling hazards for rendezvous by supply craft
are added concerns for the reactor user. The major reactor user
considerations are listed in Table 5.

CONCLUSIONS

1. The evaluation of potential military missions and papers
published on the subject indicates that the use of nuclear reactor
space power is presently not dictated prior to the 1990s. With the
lead times required, an active technology development program appears
to be prudent and desirable for both unanticipated changes in
near-term mission requirements and for large power requirements in
mission scenarios in the 1990s.

TABLE 5 Nuclear Reactor Power System User Considerations

1. Structural attachment and integration

2. Thermal input to spacecraft, heat rejection after start-up, thermal condition during launch

3. Reactor radiation dose to spacecraft components, parts selection, locating components, and dose calculations

4. Portion of reactor configuration control assigned to spacecraft, logic, instrumentation, and rationale for start-up, power reduction, temporary shutdown, and final shutdown

5. Expanded range and nuclear safety reviews, documentation, and liaison with government safety organization

6. Personnel radiation safety and shielding considerations for manned missions or manned maintenance; control of hazards associated with rendezvous by servicing spacecraft

7. Physical threat survivability associated with reactor

8. Safety considerations when fairly rapid start-up of reactor may be desirable

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2. Uncertainties in this evaluation include evolving definitions of threats, response capabilities of satellites, and subsystem design requirements.

3. Potential reactor users will hold out for nonuse. This attitude cannot be stated as a rule for all program managers but is a tendency resulting from the users' perception of the possible risks they will have to face when selecting a reactor power system, particularly those in the following areas:

o Development risk: The solar array/battery technology is older, well established, and more familiar. What availability can be counted on?

o Safety implications: The safety requirements and review procedures are well established, but feedback from the reviews by a number of reviewing agencies will provide system design "help" and less control for the program manager.

4. The reactor power system user will probably have to get a significantly high payoff for the selection as opposed to a slight

advantage over an alternative power source. This will overcome the use resistance threshold.

REFERENCES

Barthelemy, R. R., and R. Honneywell. 1982. Spacecraft power technology. Paper presented at the 17th Intersociety Energy Conversion Engineering Conference, Los Angeles, California, August 1982.

Buden, D., and J. A. Angelo, Jr. 1982. The role of nuclear reactors in future military satellites. Paper presented at the 8th DARPA Strategic Space Symposium, Naval Postgraduate School, Monterey, California, June 1982.

Koenig, D. R., and W. A. Ranken. 1982. Design options for the SP-100 thermoelectric nuclear space power plant. Paper presented at the 17th Intersociety Energy Conversion Engineering Conference, Los Angeles, California, August 1982.

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Manefkey, E. T. 1982. Space power system technology forecast. Paper presented at EIA Space Electronics Conference, Los Angeles, California, January 1982.

Malbandian, S. J., and E. P. French. 1982. Design of large, low-concentration-ratio solar arrays for low earth orbit applications. Paper presented at the 17th Intersociety Energy Conversion Engineering Conference, Los Angeles, California, August 1982.

Mullin, J. P., D. J. Flood, J. H. Ambrus, and W. R. Hudson. 1982. The NASA program in space energy conversion and research. Paper presented at the 17th Intersociety Energy Conversion Engineering Conference, Los Angeles, California, August 1982.

Powell, J. R., and T. E. Botts. 1982. Advanced reactor designs for high power generation. Paper presented at the AIAA Missile and Space Sciences Meeting, Naval Postgraduate School, Monterey, California, September 1982.
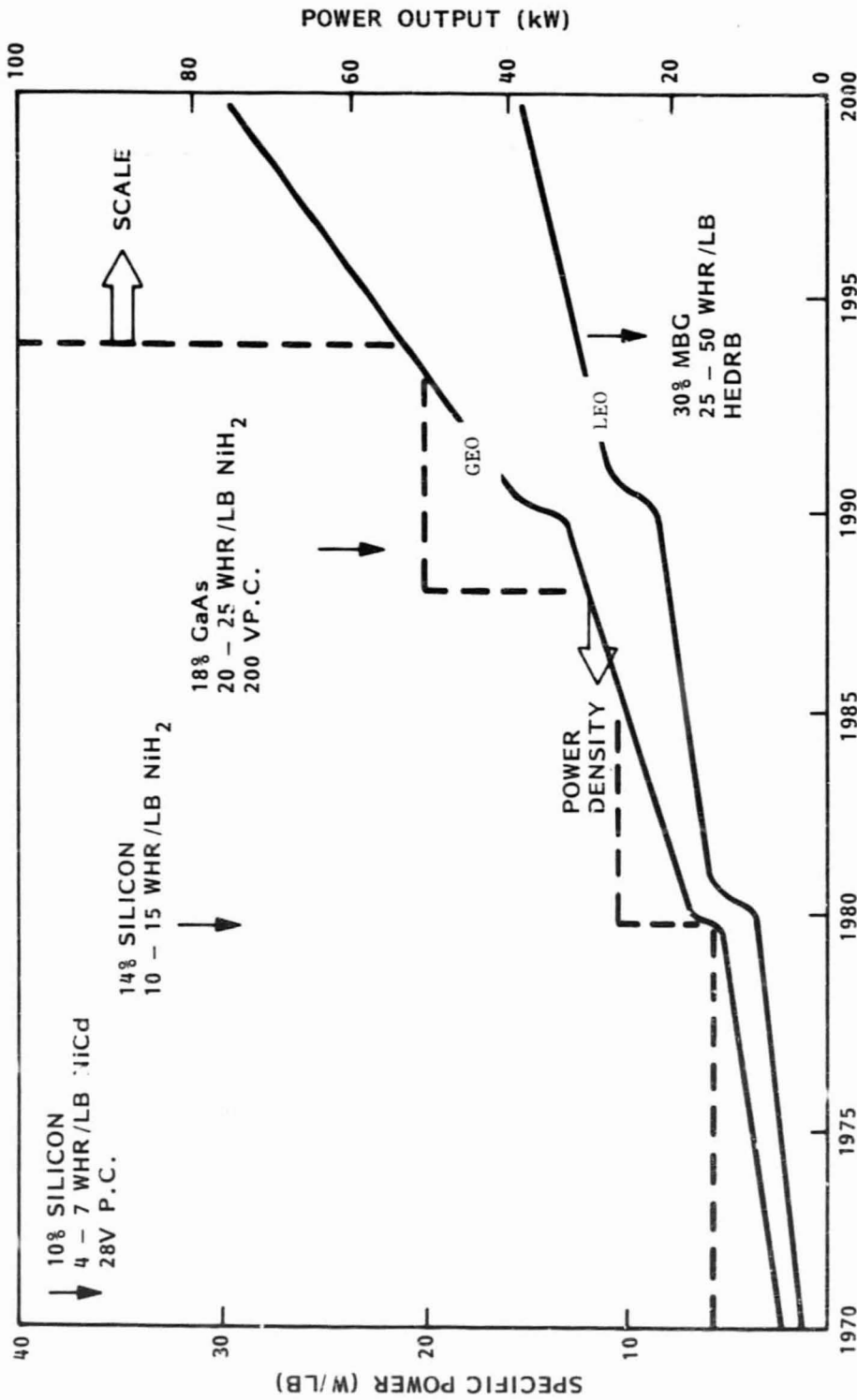
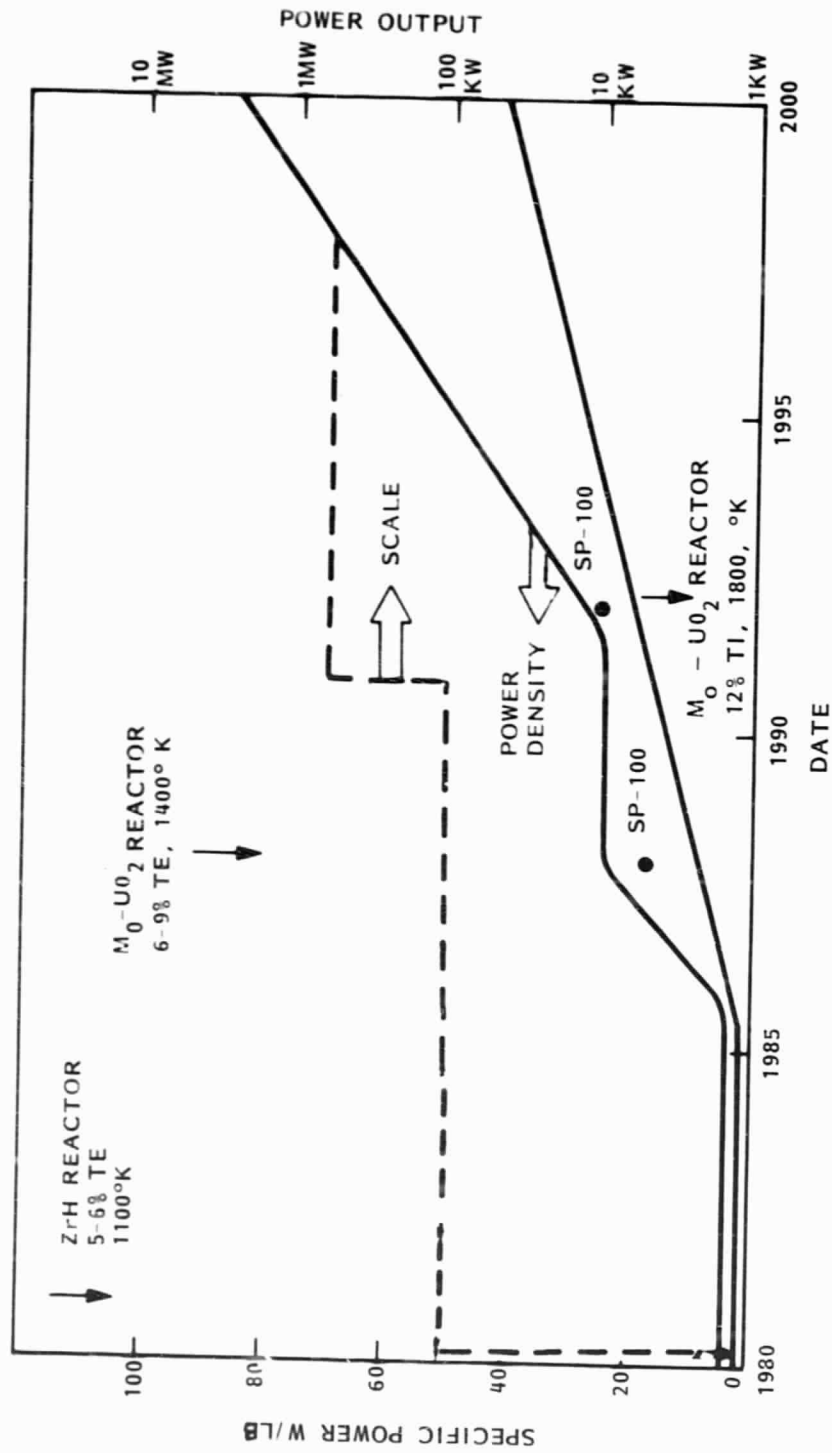FIGURE 1   Solar array power system technology.

FIGURE 2 Nuclear reactor power system technology.

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INTEGRATION OF SPACECRAFT
AND REACTOR POWER SYSTEM

EXTERNAL AND INTERNAL
INTEGRATION OF REACTOR
WITH SPACECRAFT

INTEGRATION OF SPACECRAFT (RADIATOR)
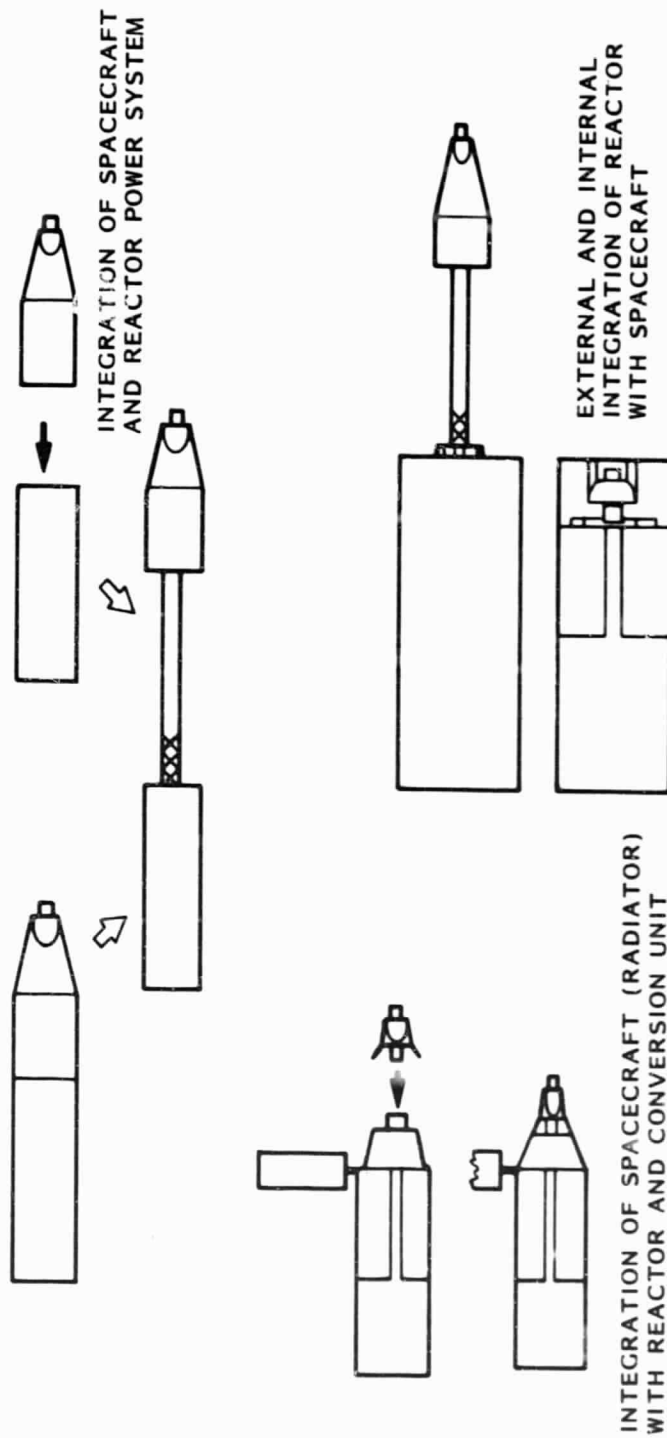WITH REACTOR AND CONVERSION UNIT

FIGURE 3   Reactor/spacecraft integration concepts.

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APPENDIXES

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APPENDIX A

PROGRAM, SYMPOSIUM ON ADVANCED COMPACT REACTOR SYSTEMS

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Symposium On

ADVANCED COMPACT REACTOR SYSTEMS

National Academy of Sciences Auditorium
Washington, D.C.
November 15-17, 1982

A number of government agencies have expressed interest in the development of lightweight, high-energy-density nuclear fission reactors with possible utility in a variety of aerospace and terrestrial applications. The National Research Council's Committee on Advanced Nuclear Systems, formed under the auspices of the Commission on Engineering and Technical Systems' Energy Engineering Board, has undertaken an assessment of the state of the art of advanced nuclear fission reactor systems. The Committee's Symposium on Advanced Compact Reactor Systems, an important element of this assessment, is intended to bring together the potential users of such systems with their developers and those who form policy affecting them. The committee will publish the proceedings of the symposium as a contribution to the literature, and will take the information and views expressed at the symposium into consideration in writing its final report.

STEERING COMMITTEE

The symposium sessions were planned and organized by the Committee on Advanced Nuclear Systems: JOHN M. DEUTCH (Chairman), HAROLD M. AGNEW, ROBERT AVERY, HERBERT GOLDSTEIN, NICHOLAS J. GRANT, HAROLD W. LEWIS, NORMAN C. RASMUSSEN, HENRY E. STONE, DAVID A. WARD, and ROBERT H. WERTHEIM.

PROGRAM

Monday, November 15, 1982

8:00 a.m. _____

Registration

8:30 a.m. _____

Welcoming remarks by JOHN M. DEUTCH, Chairman, Committee on Advanced Nuclear Systems

8:40 a.m. _____

Session 1: *Potential Mission Requirements for Compact Nuclear Reactors in Terrestrial and Aeronautical Applications* (classified)
- Introductory Remarks by HAROLD M. AGNEW, session chairman
- *Army Requirements as They Relate to Possible Use of Compact Reactor Systems*
    Speaker: J. GORDON PRATHER, Deputy for Science and Technology Systems, Office of the Deputy Assistant Secretary of the Army (Research, Development, and Systems)
- *Advanced Naval Air Vehicles: Nuclear Considerations*
    Speaker: ROBERT H. KRIDA, Manager of Advanced Scientific Studies, Naval Air Systems Command
- *Nuclear Power for Deep Basing of Missiles*
    Speaker: LT COL JAMES LEE, Chief, Nuclear Power Branch, Air Force Weapons Laboratory

10:15 a.m. _____

BREAK

10:30 a.m. _____

Session 2: *Potential Mission Requirements for Nuclear Reactors and Alternatives in Space Power and Propulsion Applications* (classified)
- Introductory Remarks by ROBERT H. WERTHEIM, session chairman

- *Outlook for Space Nuclear Power Development*
    Speaker: GORDON L. CHIPMAN, JR., Deputy Assistant Secretary of Energy
    (Breeder Reactor Programs)
- *Potential NASA Requirements for Space Nuclear Reactors*
    Speaker: JACK L. KERREBROCK, Associate Administrator for Aeronautics and
    Space Technology, National Aeronautics and Space Administration
- *Potential Military Mission Requirements for Space Nuclear Reactors*
    Speaker: COL THOMAS J. CODY, JR., Deputy Commander, Detachment 1,
    Space Division, U.S. Air Force Systems Command

12:30 p.m. _____

Lunch

1:25 p.m. _____

Session 2, continued
- *Potential Requirements for Space Nuclear Power in Future Military Missions*
    Speaker: ROBERT BARTHELEMY, Manager, Space Thrust, Air Force Wright
    Aeronautical Laboratories
- *The Military Requirement Definition Process for Space Nuclear Power Technology*
    Speaker: LCDR WILLIAM E. WRIGHT, Program Manager, Directed Energy Office,
    Defense Advanced Research Projects Agency
- *Potential Space-Borne Nuclear Power Applications: Comments and Questions from
    A Spacecraft Systems Design Standpoint*
    Speaker: JOHN A. LOVE, Special Assistant to the Director of Missions and Systems,
    TRW Space and Technology Group

3:25 p.m. _____

Break

3:40 p.m. _____

Session 2, continued
- *Space Nuclear Power Applications*
    Speaker: RICHARD V. ELMS, SEP Solar Array Project Leader, Space Systems
    Division, Lockheed Missiles and Space Co., Inc.
- *Future Military Space Power*
    Speaker: ALLEN PETERSON, Cochairman, JASON Group Panel on Future Military
    Space Power

5:00 p.m. _____

Adjournment

5:00-
6:00 p.m. _____

Reception for committee and attendees

Tuesday, November 16, 1982

8:00 a.m. _____

Registration

8:30 a.m. _____

Session 3: *Current Technological Concepts for Space Power*
- Introductory Remarks by HENRY E. STONE, session chairman
- *Radioisotope Thermal Generators and Thermoelectric Conversion*
  Speaker: GERHARD STAPFER, Group Supervisor, Nuclear Thermal to Electric Power
  Group, Jet Propulsion Laboratory
- *A New Generation of Reactors for Space Power*
  Speaker: JAY E. BOUDREAU, Deputy Associate Director for Nuclear Programing,
  Los Alamos National Laboratory
- *Some High-Temperature Reactor Technologies*
  Speaker: HAROLD J. SNYDER, Manager of Nuclear and Chemical Engineering,
  GA Technologies, Inc.

10:50 a.m. _____

Break

11:05 a.m. _____

Session 3, continued
- *Particle Bed Reactors and Related Concepts*
  Speaker: JAMES R. POWELL, Head, Reactor Systems Office, Department of Nuclear
  Energy, Brookhaven National Laboratory
- *Experience with Gas-Cooled and Liquid-Metal-Cooled High-Temperature Nuclear
  Reactor Systems*
  Speaker: RICHARD E. MORGAN, Manager, Advanced Reactor Projects, Westinghouse
  Advanced Reactors Division

12:30 p.m. _____

Lunch

1:30 p.m. _____

Session 3, continued
- *Technological Implications of SNAP Reactor Development for Future Space Nuclear
  Power Systems Activities*
  Speaker: R. V. ANDERSON, Program Development Manager, Advanced Systems,
  Energy Systems Group, Rockwell International

2:15 p.m. _____

Session 4: *Safety and Regulatory Issues Raised by Space Applications of Nuclear Reactors*
- Introductory Remarks by NORMAN C. RASMUSSEN, session chairman
- *Space Reactor Safety Strategy*
  Speaker: PAUL NORTH, Division Manager, Water Reactor Research Test Facilities,
  EG&G Idaho, Inc.
- *Regulatory Implications*
  Speaker: MANNING MUNTZING, Doub and Muntzing

3:40 p.m. _____

Break

4:20 p.m. _____

Session 4, continued
- *Safety Through Technical Integrity*
  Speaker: WILLIAM WEGNER, Associate, Basic Energy Technology Associates

- *Procedures for Securing Clearance to Launch Reactors*
    Speaker: THOMAS KERR, NASA Coordinator of the Interagency Nuclear Safety
    Review Panel

5:40 p.m. _____

Adjournment

5:45–
6:45 p.m. _____

Reception for committee and attendees

Wednesday, November 17, 1982

8:00 a.m. _____

Session 5: *Outstanding Research and Development Issues*
- Introductory Remarks by HERBERT GOLDSTEIN, session chairman
- *Status of Heat Pipe Technology*
    Speaker: WILLIAM A. RANKEN, Project Manager, Los Alamos National Laboratory
- *Refractory Metals for Nuclear Space Power and Propulsion Applications*
    Speaker: ROGER PERKINS, Senior Member, Lockheed Palo Alto
    Research Laboratory
- *High-Temperature Fuels for Advanced Nuclear Systems*
    Speaker: JAMES F. WATSON, Director, Materials and Chemistry Division,
    GA Technologies, Inc.

10:05 a.m. _____

Break

10:15 a.m. _____

Session 5, continued
- *Power Generation from Nuclear Reactors in Aerospace Applications*
    Speaker: ROBERT ENGLISH, Senior Research Associate, NASA Lewis
    Research Center
- *Neutron Physics of Reactors for Use in Space*
    Speaker: HERBERT J. C. KOUTS, Chairman, Department of Nuclear Energy,
    Brookhaven National Laboratory
- *Space Reactor Shielding*
    Speaker: DAVID BARTINE, Head, Reactor Analysis and Shielding Section,
    Engineering Physics Division, Oak Ridge National Laboratory

12:30 p.m. _____

Lunch

1:20 p.m. _____

Session 5, continued
- *Conceptual Designs for 100-Megawatt Space Radiators*
    Speaker: JOHN A. SULLIVAN, Acting Program Manager for Space Reactors,
    Los Alamos National Laboratory

2:00 p.m. _____

Adjournment

PARKING

Parking is not available at the National Academy of Sciences building. The nearest parking facilities are:

Colonial Parking on 20th Street and Virginia Avenue, N.W. (Between E and F Streets)
Columbia Plaza on 23rd and Virginia Avenue, N.W.
George Washington University's Marvin Center on H Street N.W., between 21st and 22nd Streets
George Washington University's Parking Garage on I Street between 22nd and 23rd Streets

IDENTIFICATION BADGES

Each participant will be issued one identification badge for use during all three days of the symposium. The badge will serve for entry to sessions and other functions, including lunch.

LUNCH ARRANGEMENTS

Registered participants and attendees will be served lunch in the National Academy of Sciences refectory at the times indicated on the agenda. Name badges will be required for admittance.

MESSAGE CENTER

Incoming calls to symposium participants should be directed to (202) 334-2497. Messages received will be posted on the bulletin board at the entrance to the auditorium. It will not be possible to page participants during the symposium sessions.

ADDITIONAL INFORMATION MAY BE OBTAINED FROM:

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APPENDIX B

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